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RADDA - Comparison of results of three ATWS/ATWC scenarios simulated with the help of POLCA-T and S3K/RELAP5

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Abstract

The effects of ATWS and ATWC-events with control rods failing to enter the core has been evaluated in this project. To understand the uncertainties in using modern 3D-calulation methods two different codes were used in the project. The outputs from the two code packages were compared. Within the project the used code were first evaluated against a real event, pancake core at Forsmark 3. The results give important knowledge of the core responses for such events and on how to use different code to perform such calculations. The NKS report is only one minor part of the total project. The project was sponsored by TVO, Forsmark, OKG, Ringhals, SKI besides the NKS-funding. The results could be used for PSA-studies and for deterministically safety analysis.

Key words

ATWS, ATWC, 3D-calulation comparison, uncertainties

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RADDA - Comparison of results of three ATWS/ATWC scenarios simulated with the help of POLCA-T and S3K/RELAP5

Summary

NKS, Nordic utilities (FKA, OKG, Ringhals, TVO) and SKI have supported a reseach project with the objective to increase knowledge of events in which control rods partly fail to enter into the core. Uncertainties in modern codes to perform evaluations should be evaluated.

Anticipated transients with partially failing control insertion (ATWC) have been studied with help system codes that have three dimensional neutronics models. In an ATWS analysis of a Nordic BWR of an early advanced design the control rods are inserted by the screw mechanism. In comparison of shutdown sequences, ATWC scenarios have extremely low frequency and more severe than conventional scenarios where the hydraulic scram insertion or the electro-mechanical screw insertion takes place.

Three transients, a turbine trip without bypass, two loss of feedwater with two and eight faulty open safety relief valves, were selected. It was assumed a different number of stuck control rods in each transient. The object of the studies was the Forsmark NPP unit 1. Each transient scenario was simulated with the help of POLCA-T and a coupled SIMULETE-3K/RELAP5. The results of the simulations with the two system codes are directly compared with each other. The findings are discussed.

Contents

1 Introduction

Effects of degraded control rod insertion in reactor shutdown have been investigated. Reactor power, criticality and condensation pool temperature are of main concern. The work is carried out within the framework of a project called RADDA [1, 2]. RADDA is a joint Nordic development and researched project in the area of reactor transient analysis. In particular, the project focused on understanding the effects of using three-dimensional dynamical core models coupled with advanced thermal hydraulics models. The simulation results of these postulated scenarios with failing control rod insertions are relevant for Nordic boiling water reactors.

The project work is structured into two phases. This report deals with the final part of the first phase, Phase 1. The earlier parts of Phase 1 are setting up the models including some code adjustments and developments, and a detailed validation against registrations from an actual control rod screw insertion event at Forsmark 3 1994 [3, 4]. Phase 1 included a number of transient scenarios were simulated to increase the basic understanding of cases wherein control insertion fails partially [1]. Forsmark 1 is the subject of these simulations and all subsequent Phase 2 simulations. The core is in all simulations a realistic equilibrium core of fuel having partial length fuel rods of two different lengths. As the final subtask of Phase 1, three transient scenarios are analysed more thoroughly. The selected three scenarios are simulated using two code packages of different vendors. The main results of the selected three transient scenarios are presented in this report.

The results have served three purposes:

- Firstly, the simulation results gave an overall picture of the reactor during an anticipated transient without control rods (ATWC).
- Secondly, the results were used in selecting altogether six different transient scenarios [5-10] that were analysed in Phase 2 of the RADDA project. Phase 2 studies are not further elaborated in this report.
- Thirdly, the common case simulations are used to compare the results of simulation codes with each other in order to identify possible model sensitivities and to lay base for possible uncertainty estimation.

The main purpose of this report is presentation of the common case simulations carried out by two different vendors. Discussions of applied modelling approaches and identified differences in the main results are included.

Efforts to achieve fully consistent input data between vendors were limited to two rounds of simulations by project time demands. The cases were run first with the best common input data developed and thereafter the cases were run with corrected data. Minor input mistakes were still identified after the second run. Any more rounds of simulation were not conducted based on a decision made by all the project partners. The inputs were further corrected later on for the cases ran in Phase 2 [5, 11]. Thereby, the results used in comparison evaluations and presented here are from the second round of simulations of Phase 1 with their few shortcomings.

As a whole, the obtained results from each are vendor are very good despite some remaining input discrepancy in plant system modelling, and as well some identified needs and suggestions for improved physical modelling.

2 Method, code packages and transients

The method was to carry out code-to-code comparisons. Only main results like reactor power were compared. Selected variables that present the main results are plotted in three sets of twelve figures. Conclusions are made based on the plotted results. The simplifications in modelling are discussed. The transient analysis codes used were POLCA-T and SIMULATE-3K/RELAP5. In the case of POLCA-T the neutron input data and the core burn-up data were calculated with the help of PHOENIX and POLCA-7. In the case of S3K/RELAP5, CASMO-4 and SIMULATE-3 were used. Ideally, the modelling can be done with physical best-estimate models, without any need for significant simplifications. The above mentioned validations against the Forsmark 3 event are examples of accurate modelling. This level of accuracy was not fully achieved in this work. The transient scenarios require rather advanced thermal hydraulics. One limitation in working with tedious simulations was making changes and corrections in simple modelling details. In this work and report, the discussion of modelling features is focused on code comparison.

2.1 Discussion of the applied modelling in POLCA-T and SIMULATE-3K/RELAP5

POLCA-T and SIMULATE-3K core models were taken from the data bases of an earlier study of an equilibrium core. Stationary power distributions or any stationary reactivity parameters between the two core models of the transient codes were not compared. Neither the transient codes were compared with their respective static code. Of interest are the effects of the applied thermal hydraulic options. The thermal hydraulics of a transient code is not identical with that of its respective static code. In addition to that, differences can arise from mapping of thermal hydraulics and neutronics nodes. Options like flat nodal coupling can be compared with plenum coupling. A coarse node thermal hydraulics coupling can distort the power distribution resulting in large differences in fuel bundle powers. Even in the case of compatible nodalisation between the transient code and the static code, there can appear small differences in the power distribution caused by a depletion history that is calculated with the thermal hydraulics model of the depletion code. Based on the earlier applications, it was considered that there is confidence that each model is correct and that the transient models are consistent enough with their respective static models.

In order to establish a more complete uncertainty-estimation, there remains to make comparison calculations on stationary reactivity parameters that can be of importance in core states that are encountered during transients and in the final states of simulated periods. Also shutdown margin being a more integral indicator should be analysed directly. Uncertainty analysis in this respect was not considered of highest importance for the project because both used static code systems are routinely used in ordinary in-core fuel management. Reliance on compatibility between the static and dynamical methods in question lies on correct standard application of a data transfer-link between codes in question.

Apart from truly three-dimensional best-estimate modelling, both models employed some simplifications.

In POLCA-T one quarter of the core was simulated to take advantage of the half core symmetry of the core loading and quarter core symmetry of the control rods used during depletion. In upcoming analyses, this approximation favours selection of such control rod patterns where the failing control rods are located in quarter symmetry and not for example as a cluster strictly around a position of a small shutdown margin.

In the POLCA-T calculations, the validity range of the neutron cross-sections covered only temperatures of hot conditions. It might not make much sense to analyse depressurisation transients with concurrent cooling of the reactor using the neutronics data of nominal operating conditions. However, the approximation did not seem to distort the calculations too much. Two explanations can be found. Firstly, there is a relatively small reactivity difference between cold zero-power and hot zero-power for the fuel type and equilibrium core in question. Secondly, POLCA-T uses local coolant density as a core state variable instead of local void. Nevertheless, the approximation of hot conditions is not conservative for transients where the reactor cools down so much that reactivity increases due to high coolant density and low temperature. As a secondary drawback, a cross section model valid for nominal conditions neglects the details of changing magnitude of the void coefficient when coolant density increases as a consequence of decreasing coolant temperature. The void coefficient of reactivity is no longer strongly negative at low temperatures. It should not be taken for granted that a possible extrapolation of the cross sections beyond their validity range at high coolant densities and low temperatures will be successful. In general, a cross section model should be verified at least against multiplication factors of respective lattice burnup code for the needed range of variations in core state variables.

The reactivity effect of bypass void was omitted in both models. This approximation is highly conservative for those periods of the transients where the void fraction in the bypass reaches values comparable with the void within the coolant channels. Available approaches to model the bypass void effect were merely tested. An approach is the use of an effective void as core state variable. This effective void would be a weighted average of coolant void and bypass void. In both the RELAP5 and POLCA-T models the bypass is set to be an average bypass covering the flow between fuel assemblies, the bypass flow within assemblies and the flow in the area between the core and the moderator tank wall. It would result in a substantial increase in details if the reactivity effect of radial differences in bypass void were modelled. Perhaps a limited number of lumped bypass channels would be an optimal compromise.

In SIMULATE-3K/RELAP5, the “flat” mode of nodal coupling was applied. The detailed 3-dimensional core neutronics of S3K was coupled with the coarse node model of RELAP5 T/H of 12 radial channels for active coolant flow and one for bypass flow. All the radial core channels had axially 24 nodes in both the S3K model and the RELAP5 model. The flat coupling of the codes expands the core state variables calculated by RELAP5 for each course node to values for each fine node in the S3K model without any weighing or adjusted mapping. The coarse node core state variables are fed into S3K neutronics. Therefore the model is an approximation of a true three-dimensional model.

Maximum fuel temperatures and maximum cladding temperatures calculated by S3K/RELAP5 are average pin maximums of the coarse nodes used in RELAP5. This flattens the distributions. Lower maximum values are attained compared with true maximums in the core. On the other hand, the coupled code system power distributions exhibit too high powers in the S3K nodes that are more reactive than the average coarse node. Coarse nodes force an excess of moderator into those fuel assemblies that have more power than the average power of the coarse node in question. In the fine node neutronics calculation the excess of moderator gives an incorrect increase in power. The flat coupling method results in a certain kind of time dependent homogenisation to coarse. The homogenisation takes into account insertion of control rods in detail. If very low reactor water levels are encountered, reversed core flows appear in the bypass and fuel channels in the core periphery. RELAP5 can calculate reversal flows. That was one reason for the selection of the flat coupling. It should be pointed out that the average of reactor power responded to the initiating disturbances of the transient scenarios similarly in the flat coupling simulations with SIMULATE-3K/RELAP5 and in the fine node simulations with POLCA-T.

The compared core flow rates are the average flow rates from POLCA-T and RELAP5. If there are reversed flows, positive and negative flows cancel each other in an average flow. Therefore details in the flow conditions should be checked using at least few snap-shots that are two-dimensional maps of core water and steam flow. A selection of flow channels should be compared axially, too. These comparisons were not made in this work.

Nearly identical simplified models of suppression pool temperature were set up in both codes. Temperature increase is calculated from mass and heat balances of the suppression pool water. The mass balance terms are the mass flow from the relief valves and the mass in the pool. The heat balance terms are energy coming in from the relief valve flow and energy going out due to the residual heat removal system that cools the suppression pool water and enthalpy of the pool water. A constant cooling power of 29.4 MW was used. In the RELAP5 modelling the cooling is effective while steam is being blown into the condensation pool. The rule is: $Power_{out} = min(Power_{in}, 29.4\ MW)$.

2.2 Analysed transients

Three ATWC scenarios were simulated with the help of two system analysis codes having three-dimensional modelling of the reactor core. The simulations were done for Forsmark 1 that is an internal pump BWR with fine motion control rods. In accordance with Nordic methodology for an anticipated transient without scram (ATWS), it is postulated that the hydraulic scram fails but all the control rods are inserted by means of the screw mechanism. ATWC scenarios go beyond ATWS. In an ATWC the failing control rods do not move at all. In a partial ATWC, some of the rods are inserted while other rods remain in their initial positions. Insertion of the intact rods can take place either as a fast hydraulic scram insertion or as an electro-mechanical screw insertion.

As a rule, PSA-reports include core damage frequencies for unsuccessful reactor shutdowns in combination with different initiating events. In these sequences there may be assumed different other failures in safety systems. Simulation of some shutdown sequences with the help of three dimensional reactor analysis methods can be used when setting up the success criteria for PSA. Furthermore, scenarios demonstrating unsuccessful shutdowns can provide information for PSA level 2 studies and for reviews of accident management procedures.

Three different transients were analysed. An objective was that the reactor conditions encountered in benchmarking would cover a wide range of reactor pressures. In the first transient scenario the main feedwater system functions according to the logics whereas the other two scenarios involve total loss of main feedwater. The first transient scenario was meant to represent a case with reactor pressure close to nominal operating pressure. In the second scenario the reactor pressure decreases slowly whereas it decreases rapidly in the third scenario. The three transients were:

- Case A: Turbine trip without bypass
- Case B: Loss of feedwater combined with two faulty open safety relief valves
- Case C: Loss of feedwater combined with eight faulty open safety relief valves

Different patterns of failing control rods were selected for each transient. In Case B the unaffected rods are inserted slowly by the screw mechanism. In Case A and in Case C the insertion of the unaffected rods takes place by the fast hydraulic scram.

End-of-full-power at 7500 full power hours was selected. Towards the end of the cycle the shutdown margin increases. When a disturbance happens at a time close to refuelling, fewer control rods are needed to shutdown the reactor compared with the beginning of the cycle and with the middle of the cycle. Therefore end-of-full-power results are optimistic regarding possible subcriticality.

The simulated time was 1800 s.

2.2.1 Case A, Turbine trip without bypass with 121 stuck control rods

This transient scenario represents cases in which the reactor pressure stays close to the nominal pressure. The initiating event is turbine trip without bypass. The trip signal is received at t = 1 s. The signal initiates closure of the turbine valves, opening of the safety pressure relief valves, run down of the main circulation pumps to minimum pump speed, reduction of the feedwater flow rate to 25% of the nominal value, and hydraulic insertion of the control rods as well screw insertion of the same control rods. However the screw insertion is postulated to fail completely and the hydraulic scram is postulated to be operable only for the peripheral control rods. The reactor water level starts to decrease gradually. At t = 375 s the available amount of water in the turbine condenser has been pumped into the reactor and the feedwater system stops. The feedwater temperature is ramped from 184 ˚C to 100 ˚C after all water between the high pressure pre-heater and the reactor has been pumped into the reactor. Two auxiliary feedwater pumps start to pump into the reactor at water L2, 3,1 m, 2 x 25 kg/s, and respectively two more trains of the system start to pump at L3, 2,0 m, 2 x 25 kg/s. The total effective capacity is thus is 4 x 25 kg/s. Due to the decreasing reactor water level the main circulation flow ceases gradually. After the two-phase level inside the moderator tank has fallen clearly below the water outlets of the steam separators there is very small flow of coolant returning to the downcomer.

```
- = failed rod, fully withdrawn
0 = fully inserted by fast scram


            0 0 0
        0 0 - - - 0 0
    0 0 - - - - - - - 0 0
    0 - - - - - - - - - 0
  0 - - - - - - - - - - - 0
  0 - - - - - - - - - - - 0
0 - - - - - - - - - - - - - 0
0 - - - - - - - - - - - - - 0
0 - - - - - - - - - - - - - 0
  0 - - - - - - - - - - - 0
  0 - - - - - - - - - - - 0
    0 - - - - - - - - - 0
    0 0 - - - - - - - 0 0
        0 0 - - - 0 0
            0 0 0
```

Fig. 1. Failing 121 control rods in Case A.

Table 1. Time boundary conditions in Case A.

Time [s]	**Time boundary condition or event**
1.0	Turbine stop valve closure starts and is completed in 2 seconds. Bypass valves do not open. The chosen closing time of 2 seconds is longer than in reality
1.5	Scram is triggered
1.5	Main circulation pump run down is triggered
1.5	Feedwater pump runback is triggered. FW flow rate is ramped down to 25% in 25 s. The switchover of the feedwater control mode was modelled explicitly in POLCA-T
2.5	Hydraulic scram insertion of the unaffected control rods starts. Insertion speed is 1 m/s
70	Feedwater temperature is suddenly ramped from 184 ˚C to 100 ˚C
377	The available amount of feedwater in the turbine condenser has been used. Feedwater flow rate is suddenly ramped from 25% to 0% (in 1.0 second)
1800	End of the simulation

2.2.2 Loss of feedwater combined with slowly decreasing pressure and 81 stuck control rods

In this transient scenario the initiating event is loss of feedwater. Turbine trip without bypass (TS x D) is assumed to be received 13 seconds later because of high pressure in the turbine condenser. The condenser pressure increases because it is assumed that the main coolant pumps of the turbine condenser have tripped. It is postulated that 81 control rods fail. The details of the selected transient time boundary conditions for the initial phase of the scenario do not fully comply with the actual automation of the plant. In addition, some inconsistencies in the use of time boundary conditions resulted in the fact that the simulated scenarios are not really identical with each other and that both diverge also from an expected response of the actual power plant during the first 25 seconds of simulations. These shortcomings were not considered as vital regarding the overall outcome of the later phase of the scenario and the inputs were not adjusted further for the closing runs of the transient scenario.

```
- = failed rod, fully withdrawn
0 = fully inserted in 240 s by screw


            0 0 0
        0 0 0 0 0 0 0
    0 0 0 0 0 0 0 0 0 0 0
    0 - - - - - - - - - 0
  0 0 - - - - - - - - - 0 0
  0 0 - - - - - - - - - 0 0
0 0 0 - - - - - - - - - 0 0 0
0 0 0 - - - - - - - - - 0 0 0
0 0 0 - - - - - - - - - 0 0 0
  0 0 - - - - - - - - - 0 0
  0 0 - - - - - - - - - 0 0
    0 - - - - - - - - - 0
    0 0 0 0 0 0 0 0 0 0 0
        0 0 0 0 0 0 0
            0 0 0
```

Fig. 2. Failing 81 control rods in Case B.

Table 2. Time boundary condition in Case B.

Time or set-point	**Time boundary condition or event**
2.0 s	Feedwater flow ramps down to zero in one second
15 s	Turbine stop valves start to close. Valves are closed after 1.0 seconds. Bypass valves do not open. Pressure relief valves get start order. Two valves stuck open and remain open through the transient
Water level L2	Scram is triggered at reactor water level L2 (3.1 m above top of active fuel) Electromechanical control rod insertion starts, speed 1.5 cm/s, 81 control rods in the centre of the core fail to move Two trains of the auxiliary feedwater system start with capacity 2 x 25 kg/s
Water level L3	Remaining two trains of the auxiliary feedwater system start with capacity 2 x 25 kg/s. Total capacity now 100 kg/s
Reactor pressure 12 bar	Low-pressure safety injection begins at 12 bars
1800 s	End of the simulation

2.2.3 Loss of feedwater combined with fast decreasing pressure and 15 stuck control rods

In the transient scenario Case C it was assumed that eight safety relief valves do not close until the pressure difference between the reactor and the wetwell is less than 2 bars. It was postulated that 15 control rods fail. The intact control rods are inserted by hydraulic scram. Otherwise Case C was defined identically with Case B. There occurred some input mistakes in strict interpretation of the safety relief valve logics. Therefore the relief flow rates are not identical between the two vendors. However, the obtained pressure decreases were considered to be enough similar that a general comparison was considered as possible without further corrections in the input data.

```
- = failed rod, fully withdrawn
0 = fully inserted by fast scram


                0 0 0
            0 0 0 0 0 0 0
        0 0 0 0 0 0 0 0 0 0 0
        0 0 0 0 0 0 0 0 0 0 0
      0 0 0 0 0 0 0 0 0 0 0 0 0
      0 0 0 0 0 0 0 0 0 0 0 0 0
    0 0 0 0 0 - - - - - 0 0 0 0 0
    0 0 0 0 0 - - - - - 0 0 0 0 0
    0 0 0 0 0 - - - - - 0 0 0 0 0
      0 0 0 0 0 0 0 0 0 0 0 0 0
      0 0 0 0 0 0 0 0 0 0 0 0 0
        0 0 0 0 0 0 0 0 0 0 0
        0 0 0 0 0 0 0 0 0 0 0
            0 0 0 0 0 0 0
                0 0 0
```

Fig. 3. Failing 15 control rods in Case C.

Table 3. Time boundary condition in Case C.

Time or set-point	**Time boundary condition or event**
2.0 s	Feedwater flow ramps down to zero in one second
15 s	Turbine stop valves start to close. Valves are closed after 1.0 second. Bypass valves do not open. Pressure relief valves get start signal. Eight valves stuck open
Water level L2	Scram is triggered at reactor water level L2 (3.1 m above top of active fuel) Electromechanical control rod insertion starts, speed 1.5 cm/s, 81 control rods in the centre of the core fail to move Two trains of the auxiliary feedwater system start with capacity 2 x 25 kg/s
Water level L3	Remaining two trains of the auxiliary feedwater system start with capacity 2 x 25 kg/s. Total capacity now 100 kg/s
Reactor pressure 12 bar	Low-pressure safety injection begins at 12 bars
1800 s	End of the simulation

3 Results

The results are presented in Fig. 1.1 – Fig. 3.12. Most of the figures contain one plot variable. Some figures have two or three variables of the same category. There are plotted only S3K/RELAP5 results of collapsed downcomer water level. Bypass void fraction is not included in S3K/RELAP5 plots for Case B and C. All the plotted variables are listed in Table 2.

Table 4. Plot variables used in code comparison.

Fig. in each set	**Variable**	**Notes**
1	Fission power [MW]	Includes decay heat
2	Steam dome pressure [bar]	
3	Safety relief valve flow rate [kg/s]	
4	Feedwater flow rate [kg/s] Auxiliary feedwater flow rate [kg/s] Emergency cooling flow rata [kg/s]	
5	Reactor water level [m] Collapsed downcomer level (only S3K/RELAP5) [m]	Reactor water levels are calculated by models of the actual “coarse level measurement”. Collapsed downcomer level gives information of the level in cases where the lower limit of coarse level is passed
6	Neutron multiplication factor [-]	Number of produced neutrons divided by number of absorb and lost neutrons
7	Core average void [-] Bypass average void (only POLCA-T) [-]	
8	Core inlet subcooling [˚C]	
9	Active core flow rate [kg/s] Bypass flow rate [kg/s]	
10	Core pressure drop [Pa]	

11	Maximum fuel centreline temperature [˚C] Maximum cladding temperature [˚C]	In POLCA-T values for the node average pin In S3K/RELAP5 values for large RELAP5 nodes
12	Suppression pool temperature [˚C]	

4 Analysis of the results

An expected finding from a comparison of two code systems that are widely used in different applications is that there should not appear large deviations between the codes. Large differences can appear between two nearly identical simulations if the value of some safety parameter is close to a set point of a safety system and the safety system is activated in one of the simulations.

The overall agreement between the codes was found to be good. The main transient behaviour is the same. The evaluation of the cases deals with the differences that were identified.

4.1 Evaluation of Case A

Simulation results of reactor power are in rather good agreement between the codes. S3K/RELAP5 shows oscillations in the reactor power caused by the operation of the safety relief valves. After about 800 seconds both codes predict a quasi-stationary state where the steam flow rate through relief valves is in balance with the auxiliary feedwater flow rate. The power level and the void content fall into balance. The POLCA-T simulation predicts lower reactor power for this balance. One reason for the difference is that the two code systems predict different core inlet temperatures.

Steam dome pressure in S3K/RELAP5 simulation is subject to an on-off relief flow control of the impulse operated control valves for the safety relief valves. POLCA-T shows smoother behaviour in reactor pressure because all the safety valves do not close after the first pressure relief. In the experienced POLCA-T power levels some safety valves and two pressure control valves control the reactor pressure actively throughout the transient. There is also a discrepancy in the capacities of the relief valves. A reduced capacity of 90 % was used in POLCA-T and the nominal capacity of 100 % in S3K/RELAP5. Strangely, the relief flow rate in POLCA-T simulation is larger at 1000 s compared with the S3K/RELAP5 simulation at respective moments of times. At these moments the impulse controlled pilot valves have closed the safety relief valves. This indicates that the control valve capacity is higher in POLCA-T. There are substantial periods with very constant relief flow rate in the POLCA-T simulation despite the fact that the pressure is simultaneously changing.

There is an excess of relief valve capacity. The relief flow rates of the transient correlate with reactor powers with a correction from the reactor pressures. In the beginning relief valve flow rates are different in these two calculations. This is due to the different input values for valve capacities. The capacity difference results in different modes of operation in the control logics of the pressure. In the case of the POLCA-T simulation the total relief valve flow rate is smaller. The first pressure relief takes place with a smaller derivative than in the case of S3K/RELAP5 and the electrical pilot valves start to close the main valves earlier according to closing set points 74, 73, 72 and 71 bars. The two first closing set points are reached during first 50 s. In the case of S3K/RELAP5 the set point pressures are encountered fast which gives an early closing of more main valves. Thereafter, the capacity of the open valves is no longer sufficient in the S3K/RELAP5 simulation. Individual valves steered by pressure in the steamline handle the needed pressure relief capacity. After the first main valve opening that takes place by control from electrically operated pilot valves and after the consequent electrical closure of valves, each main valve starts to operate independently in an “on-off” mode between 80 and 74 bar controlled by the impulse controlled pilot valves.

Main feedwater flow rate is a simulated variable in POLCA-T simulation and a boundary condition in S3K/RELAP5. In the POLCA-T simulation the feedwater flow rate is essentially constant during the scram control mode, which about gives 25% flow rate of nominal for a minimum time of 25 s after scram. The comparison demonstrates that consistent feedwater flow rates were used. Starts of the auxiliary feedwater system are ruled by the simulated coarse level measurements of reactor water.

Reactor water levels are similar up to 500 seconds. After that moment of time the coarse level measurements have reached the bottom of the measuring range. The lowest measured coarse range levels are different between the codes. Collapsed downcomer water level as a plot variable was available only in S3K/RELAP5 simulations.

The reactor produces fission power throughout the transient in both simulations. The power fluctuates with the pressure variations. The neutron multiplication factor of the reactor fluctuates accordingly. POLCA-T shows fluctuations in the neutron multiplication factor at the end of the simulated period with the positive reactivity comparable in magnitude with that of S3K/RELAP5. In the POLCA-T simulation the average fission power is almost smooth despite of the moments of positive reactivity.

The average core void fractions are indeed similar, but not very close to each other. After 500 seconds the average bypass void fraction is higher according to the POLCA-T simulation compared with the result from the S3K/RELAP5 simulation. It should be pointed out that the after 500 seconds the void fractions are influenced by reactor power, by core inlet subcooling and also by extraordinary thermal hydraulic conditions if reverse flows appear. In general, recirculation flow and reactor pressure have a strong effect on void, too.

The core inlet subcooling temperatures start to differ between the two calculations after the first half of the simulated time. According to POLCA-T the subcooling is zero whereas S3K/RELAP5 predicts a subcooling of 15 to 20 ºC. The reference temperature of a subcooling varies with the saturation temperature that corresponds to the actual reactor pressure between 70 and 80 bar. In POLCA-T the total flow from the downcomer to the core inlet is about 200 kg/s. There is about 100 kg/s two-phase flow of very high void content from the steam separators to the downcomer. The flow is large enough to heat up the downcomer and the cold auxiliary feedwater flow injected to the downcomer at rate 100 kg/s up to saturated condition. Therefore, one probable explanation to the different core inlet temperatures is different modelling of evaporation and condensation effects in the downcomer where cold auxiliary feedwater enters a volume with mostly steam.

The average core flow rates and bypass flow rates are in agreement. Late in the transient, the total core flows are averages of channels where the flows go either upwards or downwards. Steam and water can have different flow directions within a flow channel, too. Typically, water flows upwards in the central part of the core and downwards in the core periphery. A comparison of snap-shots of the radial flow maps would be need for more detailed comparison of the codes.

The core pressure drop is in agreement during the first 100 s. Later S3K/RELAP5 predicts a pressure drop that is about 10 % higher compared with that of POLCA-T. After 600 seconds the flow rates are so low that the main component in the pressure drop is the hydrostatic pressure of the coolant. The core void contents seem to be close to each other. However the S3K/RELPAP has 15% lower void bypass which corresponds to approximately 500 Pa. Therefore it is likely that the difference in the core pressure drops can be explained by different hydrostatic pressure drops. In POLCA-T the core pressure drop oscillates at end of the simulated period.

The maximum pellet centreline temperatures appear to be different. They are not the same variables because the POLCA-T variable is a maximum of the fine node temperatures whereas the S3K/RELAP5 variable is that for the coarse nodes. RELAP5 -value is a maximum of temperature of temperatures in 24 axial and 12 radial nodes. This discrepancy applies to the maximum cladding temperatures, too. Any specific hot-channels were not used in S3K/RELAP5 while each fuel channel in a quarter of core was simulated in POLCA-T. The initial full power fuel average temperature for the whole core was 530 ºC according to POLCA-T and 544 ºC according to S3K/RELAP5. Core heat-up can be a consequence in these kinds of transients. POLCA-T with its local modelling predicts substantial exit-dryout over large parts of the core. The maximum cladding temperatures are determined by post-dryout heat transfer. The predicted cooling is good enough to limit the increase in temperature below an acceptance criterion of 1204 ºC. The water level sinks into the core and uncovers the top in S3K/RELAP5 simulation, too. Nevertheless the fuel is still sufficiently cooled according to S3K/RELAP5.

There is at most a 25 % difference in the suppression pool temperature. Moreover, the pool temperature reaches the boiling point in the S3K/RELAP5 simulation. The only explanation should be different power levels during the transient. Both models have the same capacity for cooling of the pool water and the cooling is effective in both models during the whole transient.

4.2 Evaluation of Case B

During the beginning of the transient scenario Case B, the reactor power is in agreement but later on the results of the codes differ substantially. At 200 seconds when the unaffected peripheral control rods are nearly inserted by the screw mechanism, the reactor becomes subcritical according to POLCA-T. Thereafter the reactor is subcritical up to 1600 seconds. In S3K/RELAP5 simulation the reactor is on fission power throughout the simulated time.

The time behaviours of the reactor pressure are a consequence of the fact that two safety relief valves remain open. The pressure decreases if the generated amount of steam is less than the flow through the valves. In the opposite case the pressure increases and leads to increasing reactor power due to void collapse and consequently depending on the void coefficient speeds up the pressure increase. A pressure increase is experienced in S3K/RELAP5 results after 1000 seconds.

The reactor pressure determines the relief valve flow rate after 200 seconds when only the two failed relief valves remain open. At first the produced amount of steam is less than the capacity of the two valves at actual pressure. At the end of the S3K/RELAP5 simulation the fission power and the produced amount of steam starts to increase which leads to increasing pressure. The pressure stays below the set point of impulse opening of the safety relief valves.

The initial event of this scenario is loss of feedwater. Therefore main feedwater flow rate remains at zero throughout the simulation with the exception of the first seconds of the transient where the feedwater flow is ramped to zero in 1 second in POLCA-T and in 2 seconds in S3K/RELAP5 simulation. The events of possible interest for comparison are limited to start of auxiliary feedwater and start of low-pressure safety cooling injection. In the POLCA-T simulation the lowest reactor pressure is slightly above 12 bars. At pressures below 12 bars, the low-pressure safety cooling system would start to deliver water into the reactor, too. Regarding conclusions that may be drawn from the simulated scenario, it is worth to notice that a possible start of the low-pressure safety injection would have changed the character of the scenario.

The consistency of the compared reactor water levels is influenced by the fact that the reactor power levels start to diverge from each other. The water level decreases are in a truly good agreement at the beginning of the transient. In the POLCA-T simulation the measured downcomer water level does not fall below the lower limit of the coarse range measurement at any time although margin to the lower limit is small. The coarse level measurement in the S3K/RELAP5 simulation falls below the bottom of the measurement range. Around 200 seconds there develops a considerable discrepancy between the coarse level of POLCA-T results and the collapsed downcomer water level of S3K/RELAP5 results. The time is not enough for the reactor vessel coolant inventories to deviate that much. The different power levels and thereby different amounts of steam out from the reactor vessel is a contributing factor in the observed different water levels. If the void conditions are different under the lower nozzle of coarse range measurements, different pressures may be experienced in wide range measurement nozzles because void below the lower nozzle lifts the two-phase level. Core flows become very small but possible effects of flow on simulated level measurement has not been checked. The pressure difference is interpreted as hydrostatic pressure that is converted to water level. In these conditions the two-phase level will be higher than the collapsed water levels. The simulated measurements reflect collapsed levels. The codes use different models for void degradation of the pump head. Results on the main circulation flow of POLCA-T point to the fact that loss of main circulation pump head takes place in a periodic manner. The pumps are self-ventilating because after a loss of the pump head the steam can more easily flow upwards. U-tube oscillations can be a driver of the periodic changes in main circulation flow, too. The oscillatory variations in the main circulation flow do not give any notable changes in the downcomer level in the simulated coarse level measurement of POLCA-T.

The reactivity is very different from the moment that POLCA-T predicts a subcritical reactor state at 200 seconds at what time the screw insertion is nearly completed. The reactivity differs by 5000 pcm between the calculations. If the void coefficient of reactivity is between –120 pcm/void% and –180 pcm/void% in these highly voided cores, the average void should be about 30 %-units less in the critical part of the S3K/RELAP5 core. The average voids differ in the expected direction by 5 – 9 %-units. It looks like the difference in reactivity is not explained only by differences in the core average void. A possible explanation to the large difference in the core criticality is the coarse nodalisation used in RELAP5. The flow channels of the core are lumped into 12 thermal hydraulic channels in three radial zones and azimuthally in each core quadrant. The pattern of the failing rods is quadrant symmetrical and the half symmetrical core loading is approximately quadrant symmetrical. Therefore the RELAP5 model has essentially three radial coarse nodes. The pattern of the failing rods is such that roughly 1/3 of the fuel channels in the area of the middle zone do not have a control rod present and 2/3 of the fuel channels do have a control rod inserted between the channels. The coolant flows of channels not having a control rod should become overestimated in RELAP5 nodalisation. A comparison of snap-shots at suitable moments of time would be needed. Later in the transient the reactor pressures are different. Therefore also the moderator temperatures are different and the temperature effect on reactivity should be taken into account. Regarding POLCA-T, the set of neutron cross sections is valid for nominal operating conditions and low temperatures lead to an extrapolation of the neutronics data.

Average voids do not agree in detail but the transients are too different for detailed conclusions. In the POLCA-T simulation voiding is mainly caused by depressurization, whereas in the S3K/RELAP5 simulations, the reactor power contributes to void significantly. In addition, the opening and closing of safety relief valves differ in the early phase of the simulations. Therefore the voids can be compared with each other only imprecisely.

The core inlet subcooling temperatures are in fine agreement in the early phase of the transient. After that the decreasing pressure keeps inlet subcooling close to zero in both simulations. Toward the end of the simulated period subcooling is observed for both codes although the transients have diverged from each other so much that the values should not any longer be compared with each other.

The core flow rates and bypass flow rates agree well during those phases when the conditions in the reactor vessel are similar. As a discrepancy, POLCA-T predicts peaks in the flow rates between 200 – 700 seconds whereas S3K/RELAP5 flows are smooth.

The maximum fuel centreline and cladding temperatures follow the reactor power and the pressure as expected. One small detail is an increase of the pellet centreline temperature in the POLCA-T simulation at time 100 seconds. It is likely that this increase is caused by increased moderation and thereby higher fission power above the front of the unaffected control rods that enter the core by the screw insertion. Dry out is not predicted. However, the validity of the applied heat transfer model at low pressure is not assessed.

The suppression pool water temperature is an integrated measure of produced energy and energy from cooling the pool water and heat initially stored in the structures of the reactor. The differences in the fission powers result very different time behaviour of the pool temperatures, too.

In Case B, poor coincidence is found regarding reactivity in the simulations, thereby resulting in two different scenarios.

4.3 Evaluation of Case C

In Case C the reactor becomes subcritical by hydraulic insertion of the unaffected control rods. There are few differences in the applied time boundary conditions of the transient during the first 25 seconds, but these deviations have only small effect later on. At the end of the simulated times there is a large relative difference in reactor powers. However, the explanation is simple. The reactor is shutdown and produces residual power only. In the S3K/RELAP5 simulation the calculation of residual power is based on an ANSI/ANS model whereas a constant residual power is used in the POLCA-T simulation.

The reactor pressures decrease in similar manner but with a time shift. The points of time for start of ADS are different.

The safety relief valves are operated differently during the beginning of the transient. Fewer valves are initially opened in the POLCA-T simulation compared with Case B and compared with the S3K/RELAP5 simulation. In addition to that, the exploited capacities per valve are different. Deduced 90 % capacity of the nominal capacity is used in the POLCA-T simulation and 100 % capacity in the S3K/RELAP5 simulation.

The starting times for the auxiliary feedwater system (327) are somewhat different. Also the low-pressure safety injection system (323) starts to pump into the reactor in both simulations. In the S3K/RELAP5 simulation 4 x 323 are used whereas in the POLCA-T simulation 1 x 323 is used.

In the S3K/RELAP5 simulation the coarse level of reactor water sinks more rapidly in the beginning of the transient. A contributing factor could be the higher flow rate in the relief valves.

The reactivity is similar in both simulations. In the S3K/RELAP5 model there are in the inner most part of the core mapped together about equally many flow channels with a control rod inserted and without a control rod. Mapping works well in this transient because the thermal-hydraulics becomes similar for controlled and uncontrolled areas since there are not any large differences in fuel bundles powers. The reactor is subcritical and the fuel assemblies are producing only decay heat. Due to large differences in low-pressure emergency core cooling flow rates the void fades away earlier in S3K/RELAP5 results that are calculated with all the four trains of the system in action. Vanishing void increases reactivity in both simulations. The limited validity range of the neutronics data used in the POLCA-T model should be kept in mind when comparing the reactivity. The reasonable agreement in the core reactivity may be resulting from an extrapolation outside of the validity range that happens to be feasible in this case and by the fact that the isothermal temperature effect of reactivity is small.

The average core void fractions look similar during one period that starts after the initial opening of the relief valves and ends before the start of the low-pressure safety injection.

The core inlet subcooling values are in agreement. The subcooling goes to zero for both cases during the blow down.

The active core flow and the bypass flow rates show different behaviour between 100 to 300 seconds. The POLCA-T simulation shows slightly higher circulation flow rates after the initial phase of the simulation.

The core pressure drops differ in accordance with core flow rates. At 700 seconds the cold water from the safe injection systems sets on heavy oscillations in the POLCA-T simulation as can be seen also in the core and bypass flows. The core inlet subcooling starts to increase at the same moment of time.

The maximum fuel centreline temperature in the POLCA-T simulation is higher throughout the simulation. Since POLCA-T uses a constant decay heat and on the other hand S3K/RELAP5 values are coarse node averages, it is difficult to judge if the true pellet-to-coolant heat transfers differ. An evaluation would require a close look into the models and to local results. Cladding temperatures follow coolant temperatures.

The condensation pool temperatures are similar. The low-pressure emergency core cooling system uses the condensation pool as a water source. Heat up of the pool water increases the temperature of emergency cooling water that is not taken into account. Thus the pool heat-up temperatures do not illustrate real circumstances. The temperature increase illustrates the buffer capacity of the pool. The pool receives the energy stored in reactor water, initial steam, reactor vessel and vessel internal metal structures.

5 Conclusions

5.1 Agreements in the results

It is demonstrated that it is possible to achieve similar results by the applied codes. There are phases in the different scenarios where the agreement is very good. For a better overall agreement between the results some simple input corrections are needed. In general, an agreement between two established codes is a rather expected finding.

5.2 Disagreements in the results

Many of the disagreements have rather straightforward explanations as discussed earlier. Some of the observed disagreements can be debated closer only with new simulations. There are disagreements that may originate from parameterization of the two-group neutron cross section. Coupling of the S3K neutronics and the RELAP5 thermal-hydraulics seems to be a source of disagreements. One significant and more obvious difference was found, too. Different core inlet temperatures were observed when the reactor water level was very low and the main circulation flow had practically ceased.

Limited differences in core inlet temperatures may look rather unimportant but underling physical explanations can have an important role in a variety of quasi-static reactor conditions with very low main circulation flow. The core inlet temperature differences are presumably caused by different heat-ups of auxiliary feedwater when it enters the downcomer. At low water levels inlet water enters to the steam phase and water is heated up by steam condensing. The magnitude of the steam condensing heat-up can influence a scenario substantially as saturation at core inlet reduces the power level of a critical reactor. On the other hand unequal temperature distribution within the water phase can influence the allocation of coolant between the downcomer and the moderator tank. Measured downcomer water levels can be affected, too. Ideally, an in depth investigation of the downcomer water heat-up should consider a variety of transients and assess different modelling assumptions. Modelling investigations should include effects of heat from the vessel walls, water in the control rod drive tubes, flow of purge water through control rod drives, the speed of rising bubbles, and the sensitivity of these factors and other possible factors that may play a role in modelling sequences with a two phase level no longer reaching the outlets of the steam separator. In this study, it was identified that condensation of steam in the downcomer is one factor contributing to reactor temperature. The scope of an investigation should include sequences that lead to collection of cold water at the bottom of the pressure vessel, too. The modelling assumptions needed to create cold-water sequences should be compared with assumptions that result in uniform heat-up of the reactor coolant up to saturation.

5.3 Recommendations for using 3D-codes for ATWS/ATWC-scenarios

The obtained results of the scenarios studied in this report are a verification that an ATWC can progress into a quasi-static stage during which the reactor power depends on the feedwater flow rate rather than on the number of failed control rods. Dynamic three-dimensional power distribution simulations supplemented with possible static calculation can be used in identifying scenarios where stable subcritical conditions are encountered with margin. A vital reactor safety concern of ATWC scenarios is the risk that the scenario progress into a fast developing severe reactor accident with high activity release in an early loss of the containment integrity. In that light, the simulations presented here are examples of how 3D-codes can be used to understand the early phase of PSA level 2 ATWC sequences [12]. The consequences of sequences leading to core melt are studied with the help of severe accident analysis codes. The results of detailed 3D-codes in their range of validity can be used to tune some correlation and models of severe accident analysis methods, including models for fission power. If a code with a three-dimensional core model has feasible fuel heat transfer models, the magnitude of an ongoing cladding oxidation could be estimated.

Detailed and accurate 3D-simulations are most useful when studying such partial or full ATWC-scenarios that are in between clearly successful shutdowns and fast developing severe accidents. The progression of the scenarios are depending more on the position of the rods that fail than on the number of the rods. In this work hypothetical cases with zones of missing control rods were studied. It is recommended to study rods failing in random patterns as well as distributed uniformly in accordance to the four electrical sub-systems. If the number of missing rods in the zone exceeds certain values the reactor will stay critical or gain recriticality. In such ATWC-scenarios, the reactor power, after an initial phase, is primarily determined by the auxiliary feedwater flow rate. Analyses on the recommended scattered rod cases are seen as need for deeper understanding of the quite large, and thereby important, category of the cases that have ”intermediate consequences”.

5.4 Other aspects concerning ATWS/ATWC evaluation

Out of a variety of considerations, here three aspects related to transient simulations are elaborated on.

Different factors contributing to a risk of rapid power excursions during the quasi-static state of ATWC and during the refill phase should be studied.

Mass and temperature distributions of the coolant within the reactor pressure vessel were found to be sensitive to both modelling and to an ATWC-sequence in these cases. Thermal hydraulics should be assessed further. For example knowledge on measured reactor water level as an indicator in abnormal conditions could benefit if such further studies are made.

An area where improvements should be targeted is more accurate use of neutronics data. Void in bypass should be added to the core state variables. As a goal, the fine accuracy of lattice neutron transport code results should not be disturbed more than what is unavoidable when using few-group neutron cross-sections in transient codes.

References

[1] *RADDA Seminar, February 2-3 2006*, Forsmark Kraftgrupp CD containing minutes of the seminar and the presentations, 2006

[2] *RADDA – Final Seminar 2007-02-15*, Forsmark Kraftgrupp document containing minutes of the seminar and the presentations, 2007

[3] *POLCA-T Validation against Forsmark 3 Control Rod Insertion Event 1994-06-15,* Westinghouse Report, SES 05-282

[4] *Validation of a coupled code system SIMULATE-3K/RELAP5 against a recoded transient in Forsmark-3 with slow control rod insertion*, Studsvik Scandpower, SSP-06-201

[5] *RADDA Project Phase 2, Fast Closure of the Main Steam Line Isolation Valves with Pump Runback Versus Number of Failing Control Rods*, Westinghouse Report, SES 07-046

[6] *RADDA Project Phase 2, Heat up of the condensation pool versus number of failing control rods*, Westinghouse Report, SES 07-051

[7] *RADDA Project Phase 2, Sensitivity of feedwater level control on type Case A event*, Westinghouse Report, SES 07-054

[8] *RADDA Project Phase2, Comparison of Results at BOC, MOC and EOFP core states*, Studsvik Scandpower, SSP-07/201

[9] *RADDA Project Phase 2, Failure of 3 out of 4 Electrical Trains for the Mechanical Control Rod Insertion*, Studsvik Scandpower 2007, SSP-07/202

[10] *RADDA Project Phase 2, Cold Water Transient with Failed Control Rod Insertion*, Studsvik Scandpower 2007, SSP-07/203

[11] *Documentation of the S3K/RELPA5 Model Used in the RADDA Project*, Studsvik Scandpower 2006, SS-06/212

[12] *RADDA – Preliminär slutrapport, Redovisning av viktigare slutsatser från projektet*, Forsmark Kraftgrupp AB, Report FT-2007-0617 (in Swedish)

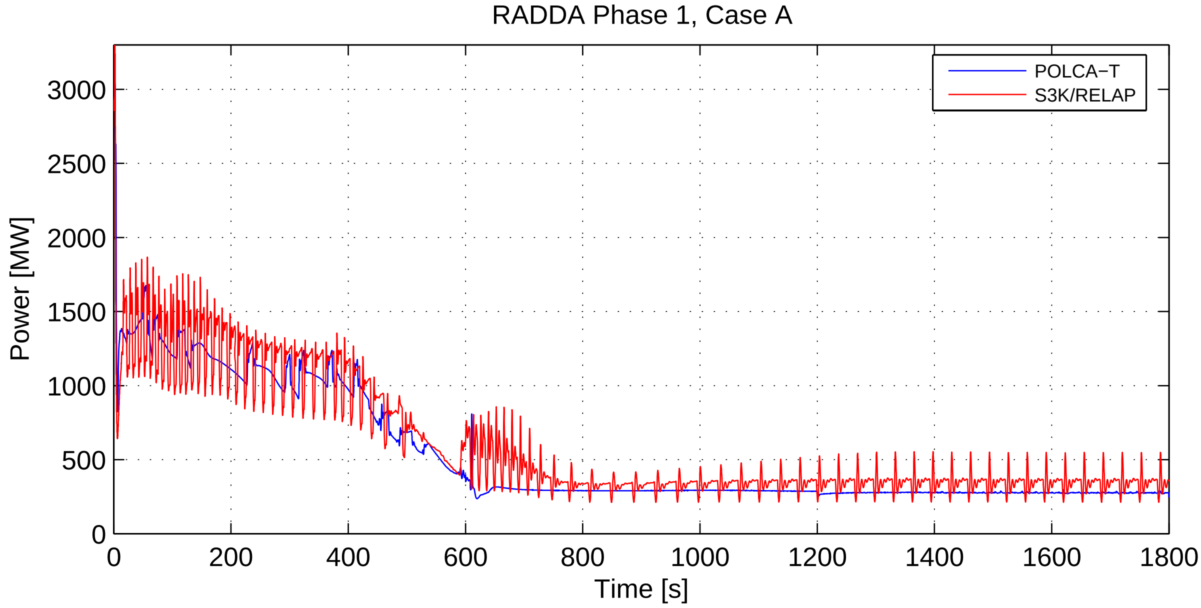

Fig. 1.1. Reactor powers in Case A, turbine trip without bypass, fast hydraulic scram, 121 failing control rods.

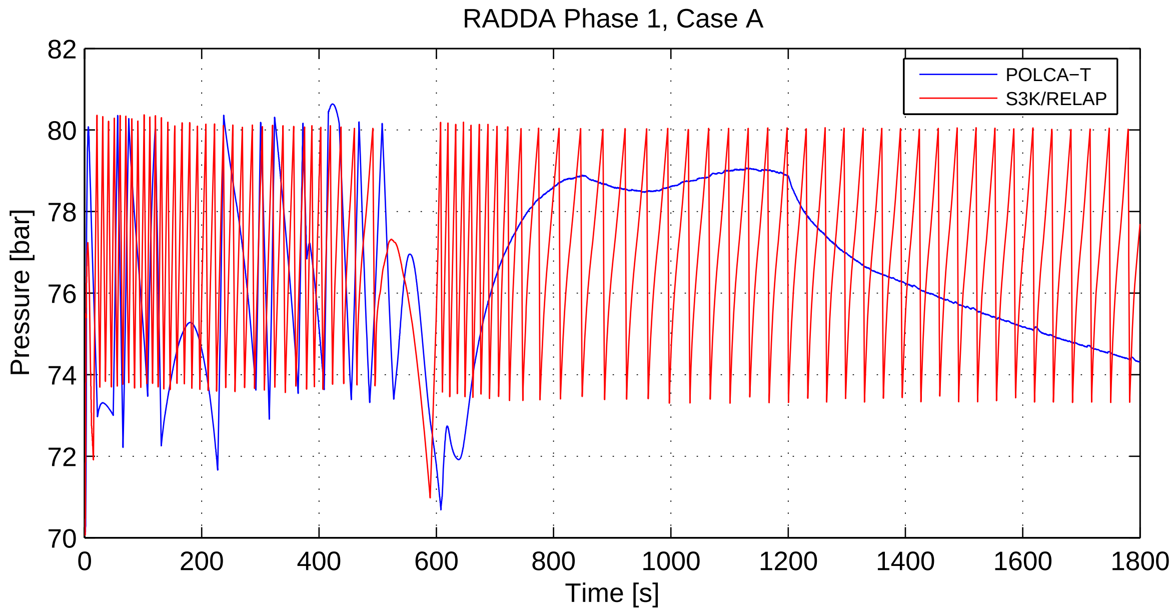

Fig. 1.2. Reactor pressures in Case A, turbine trip without bypass, fast hydraulic scram, failing 121 control rods.

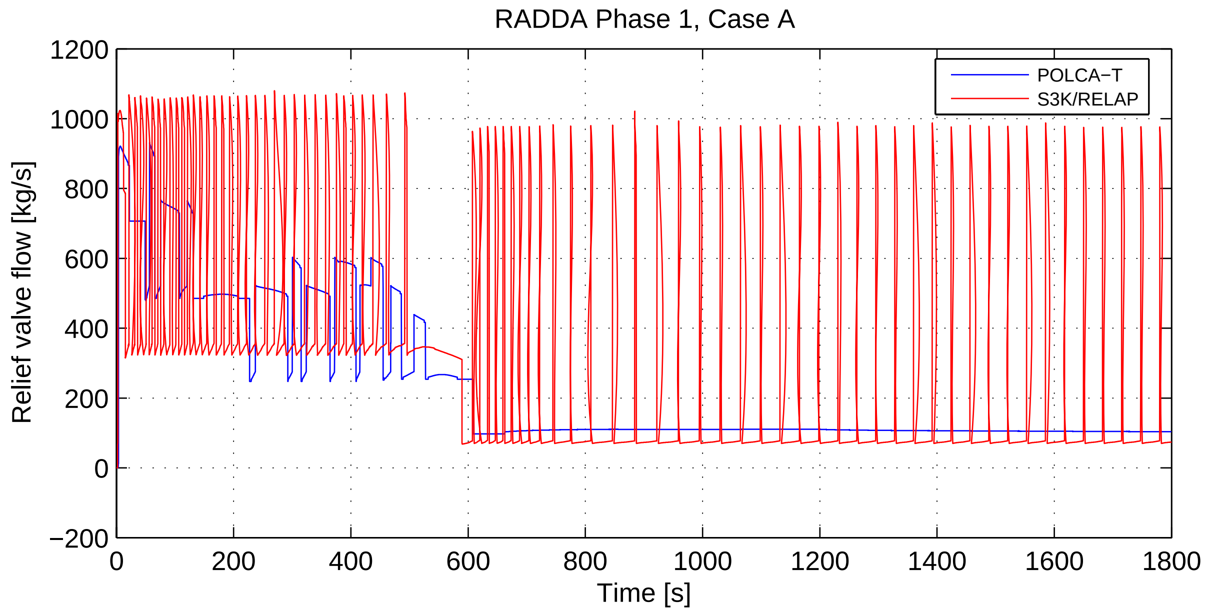

Fig. 1.3 Pressure relief valve flow rates in Case A, turbine trip without bypass, fast hydraulic scram, 121 failing control rods.

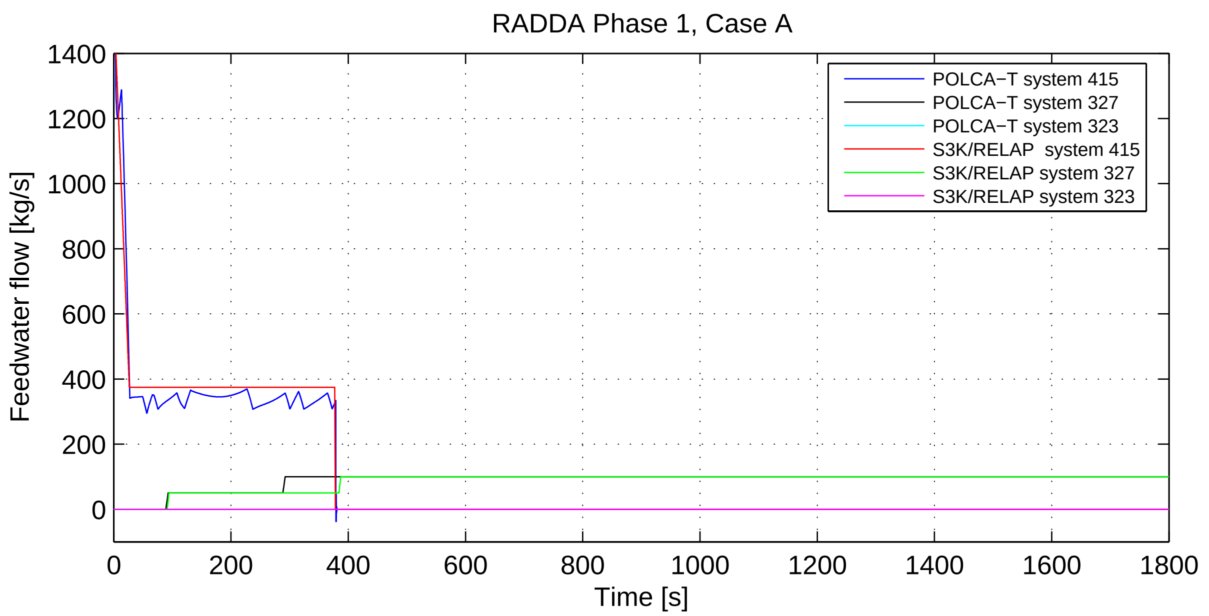

Fig. 1.4 Feedwater flow rates into the reactor pressure vessel from the feedwater system 415, the auxiliary high pressure feedwater system 327 and the low-pressure emergency core cooling system 323 in Case A, turbine trip without bypass, fast hydraulic scram, 121 failing control rods.

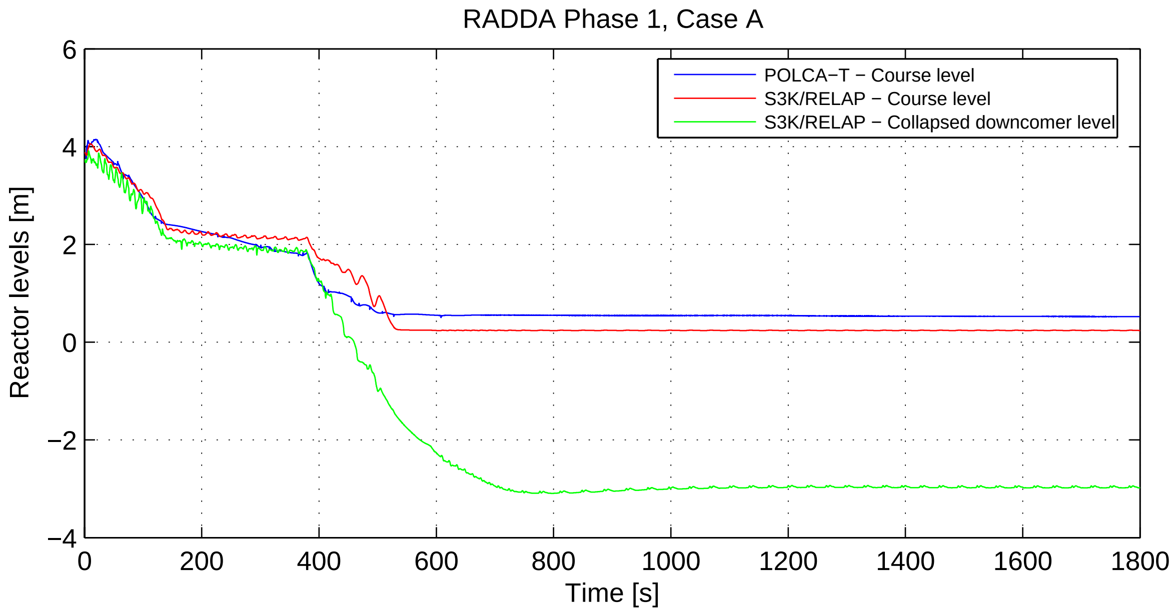

Fig. 1.5. Reactor water level according to the coarse range measurement and the collapsed downcomer water level from S3K/RELAP5 in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

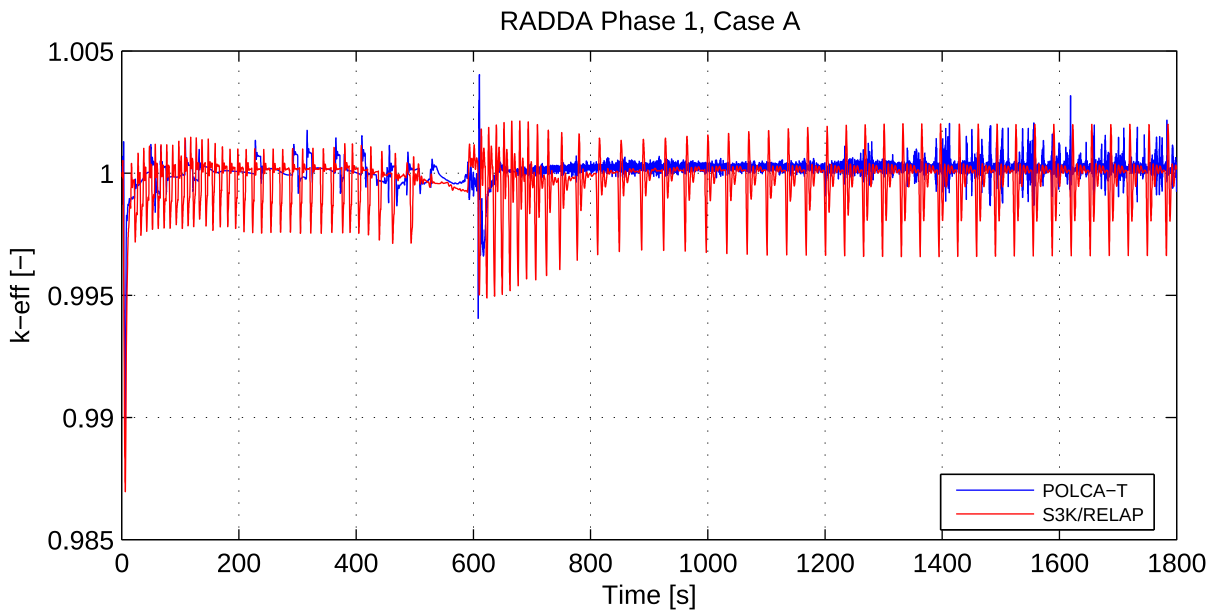

Fig. 1.6. Neutron multiplication factor as ratio of produced neutrons and absorbed + lost neutrons at actual moment of time in Case A, turbine trip without bypass, fast hydraulic scram, 121 failing control rods.

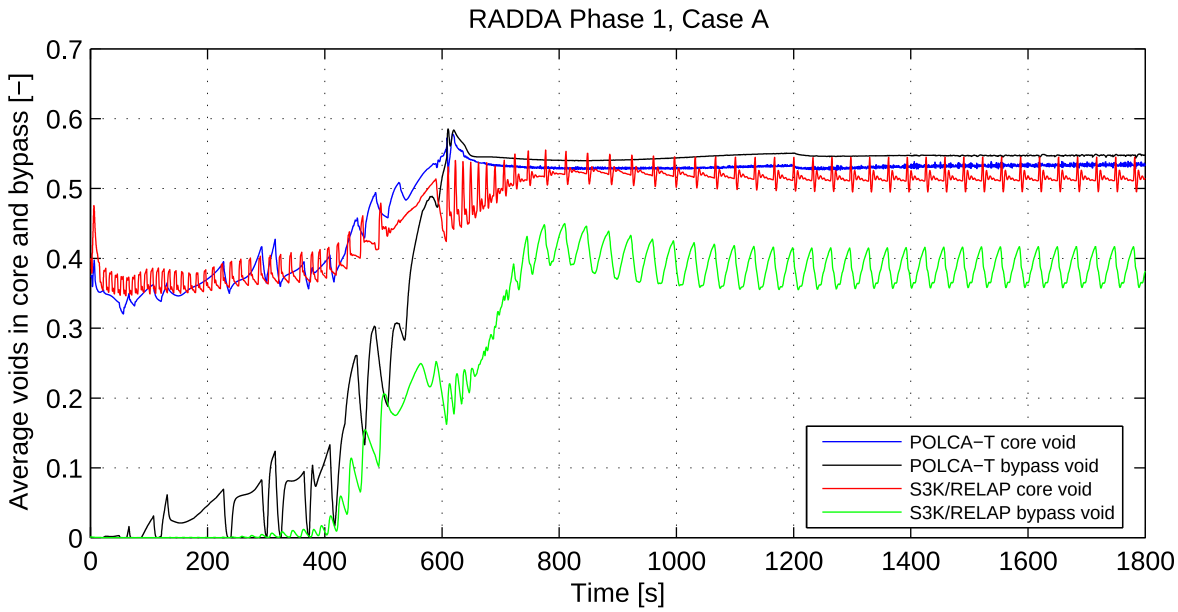

Fig. 1.7. Average void fraction of active coolant flow and of bypass flow in Case A, turbine trip without bypass, 121 failing control rods.

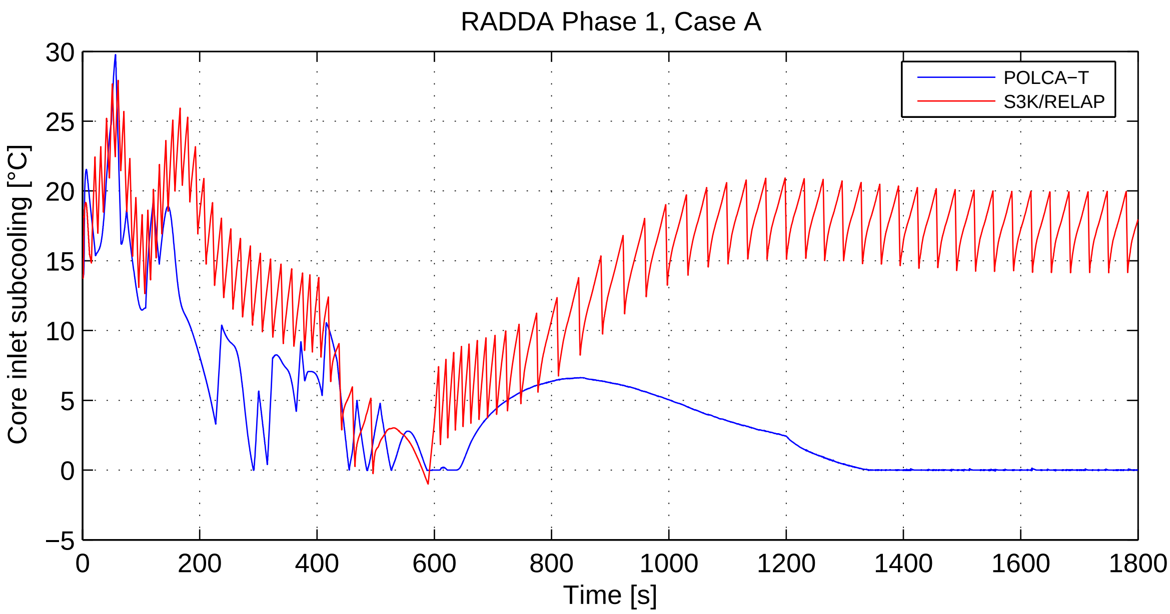

Fig. 1.8. Coolant subcooling at the core inlet in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

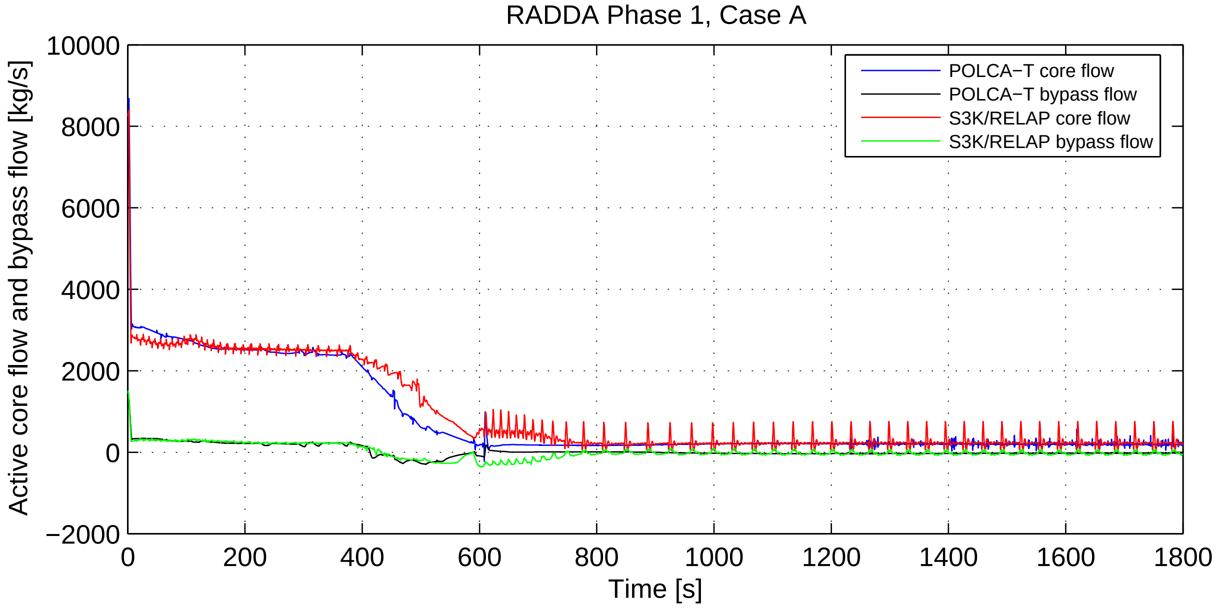

Fig. 1.9. Core coolant flow rate within fuel channels (active flow) and outside fuel channels (bypass flow) in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

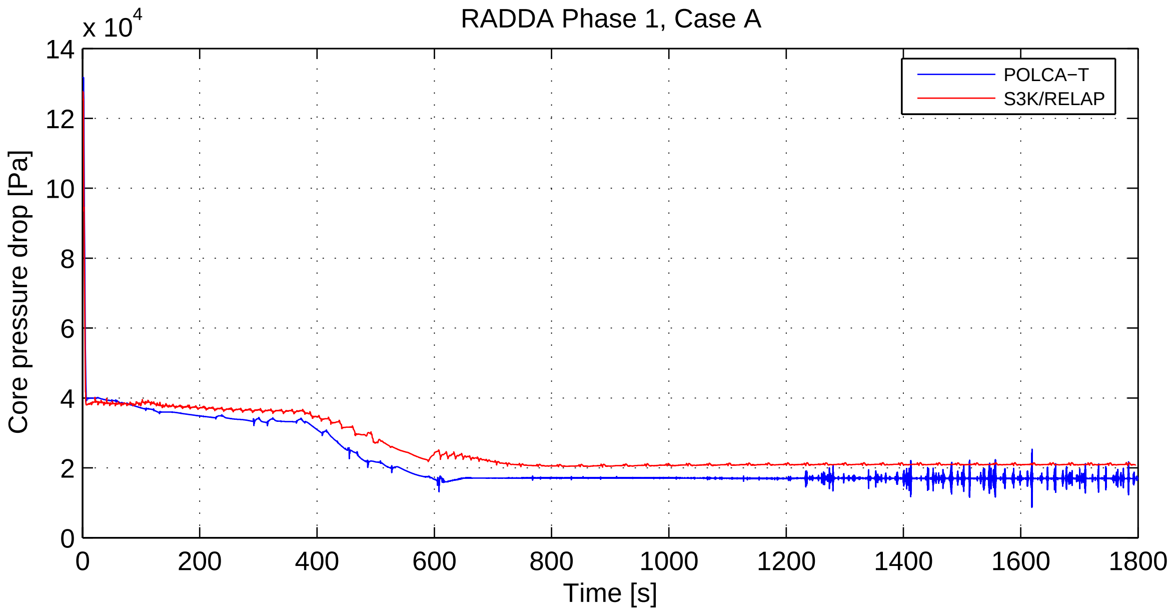

Fig. 1.10. Core pressure drop in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

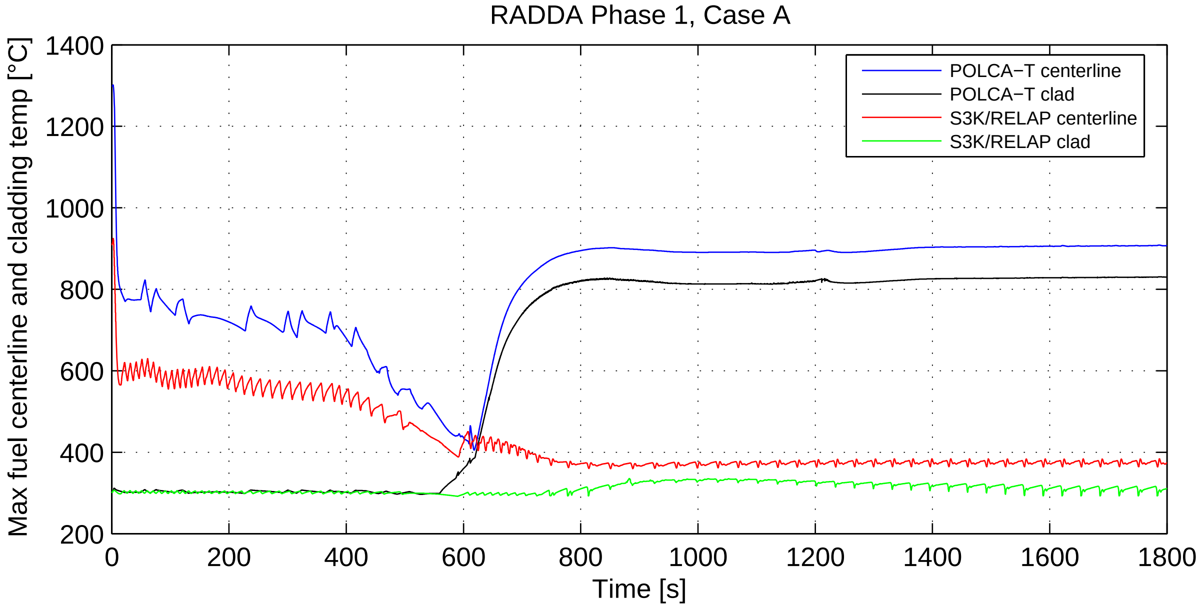

Fig. 1.11. Maximum fuel centreline and cladding temperature, for S3K/RELAP5 maximum coarse node temperature in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

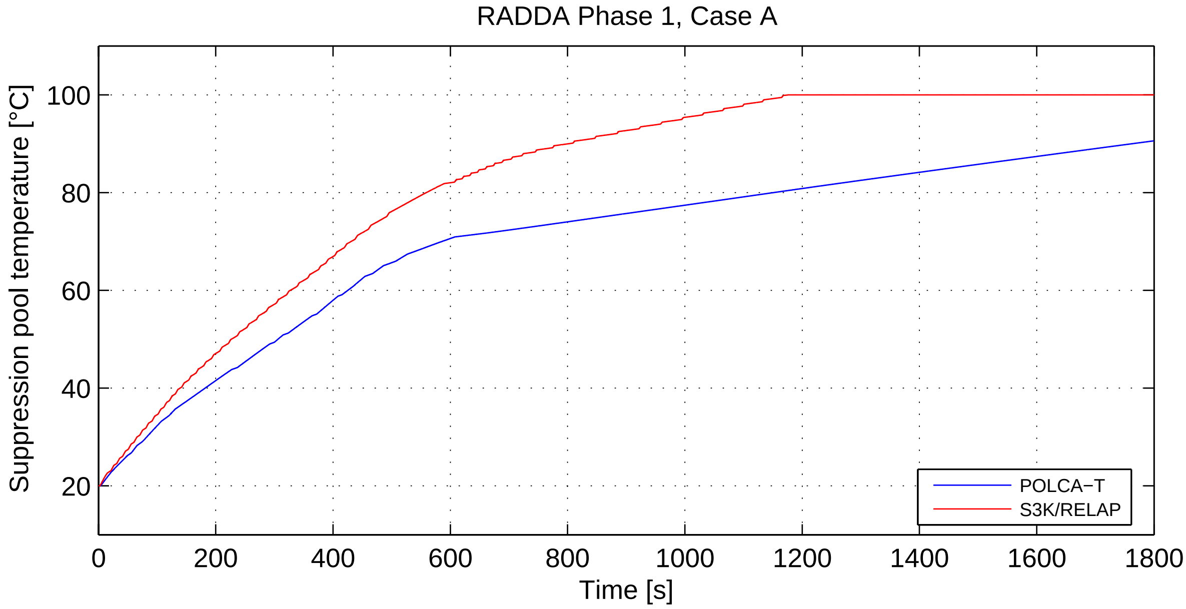

Fig. 1.12. Simple model suppression pool temperatures in Case A, turbine trip without bypass, hydraulic scram, 121 failing control rods.

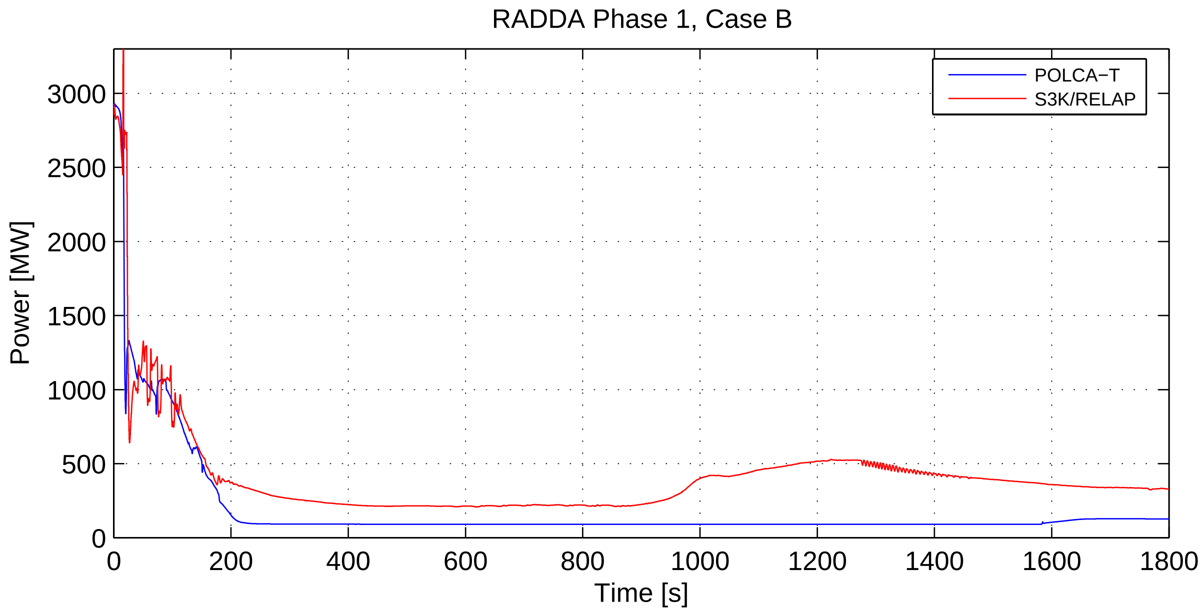

Fig. 2.1. Reactor power in Case B, loss of feedwater, 2 faulty open safety relief valves, hydraulic scram, screw insertion with 81 failing control rods.

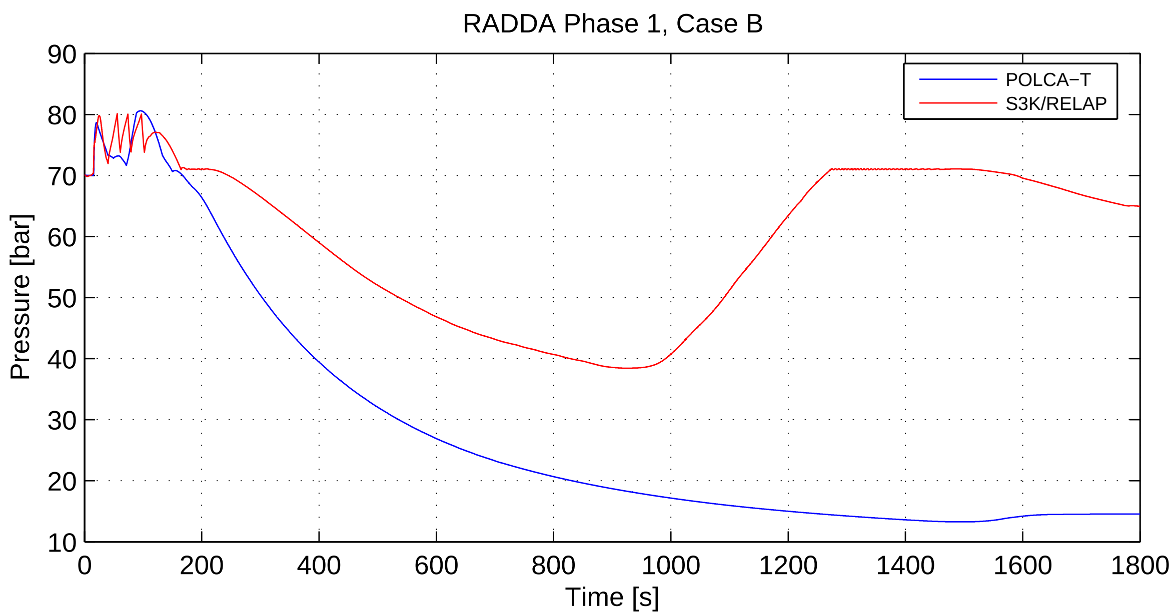

Fig. 2.2. Reactor pressure in Case B, loss of feedwater, 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

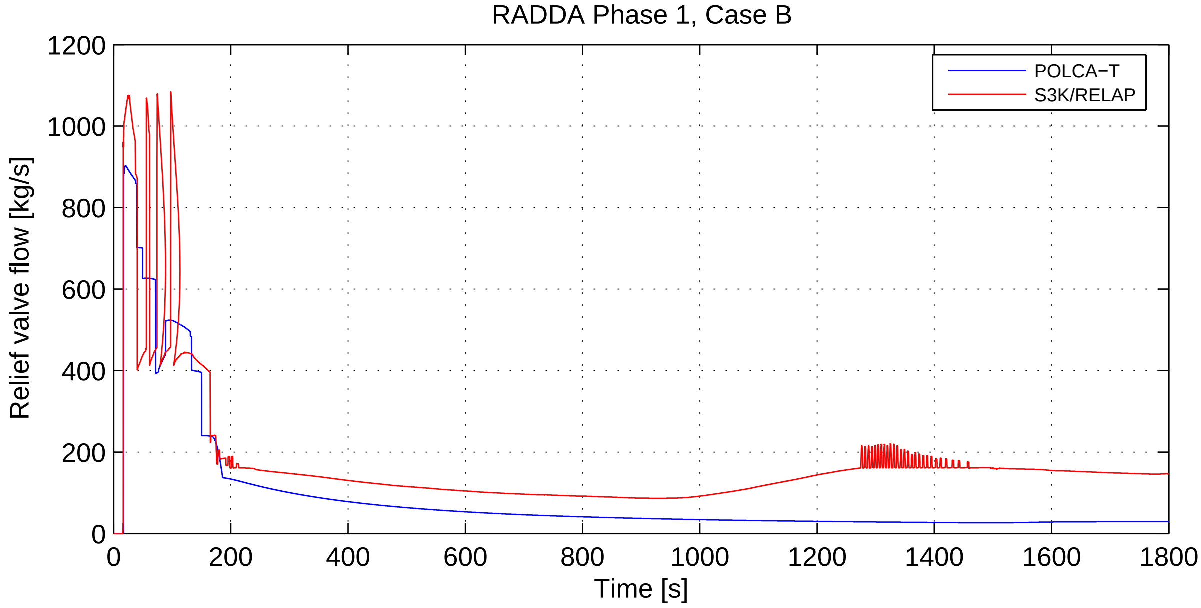

Fig. 2.3. Pressure relief valve flow rate in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

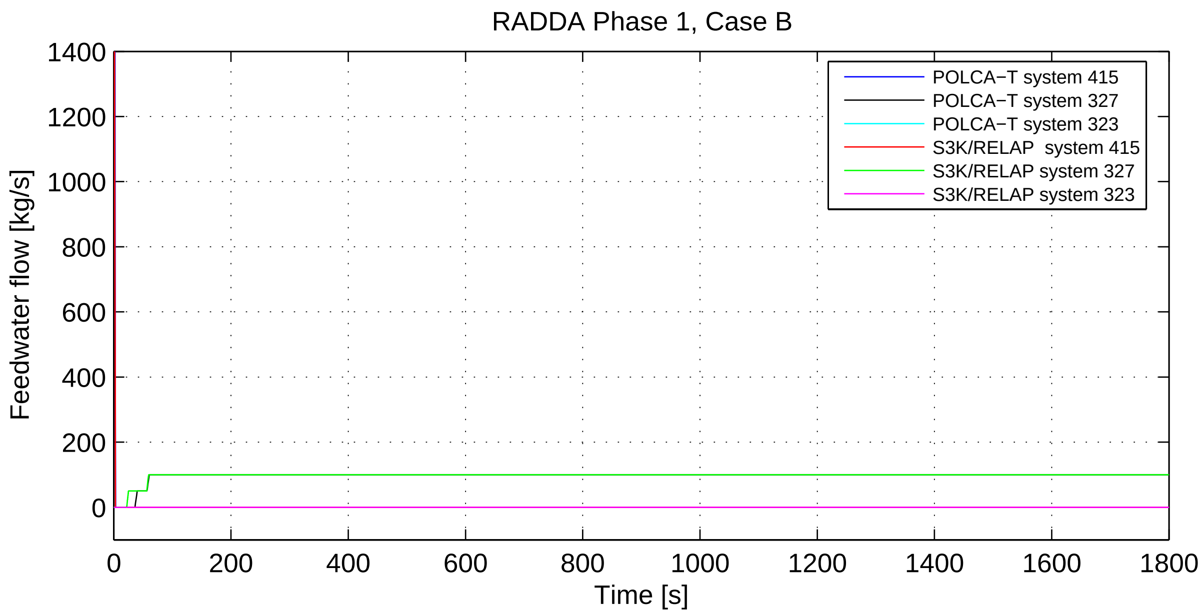

Fig. 2.4. Feedwater flow rates into the reactor pressure vessel from the feedwater system 415, the auxiliary high pressure feedwater system 327 and the low-pressure emergency core cooling system 323 in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram of control rods, screw insertion with 81 failing control rods.

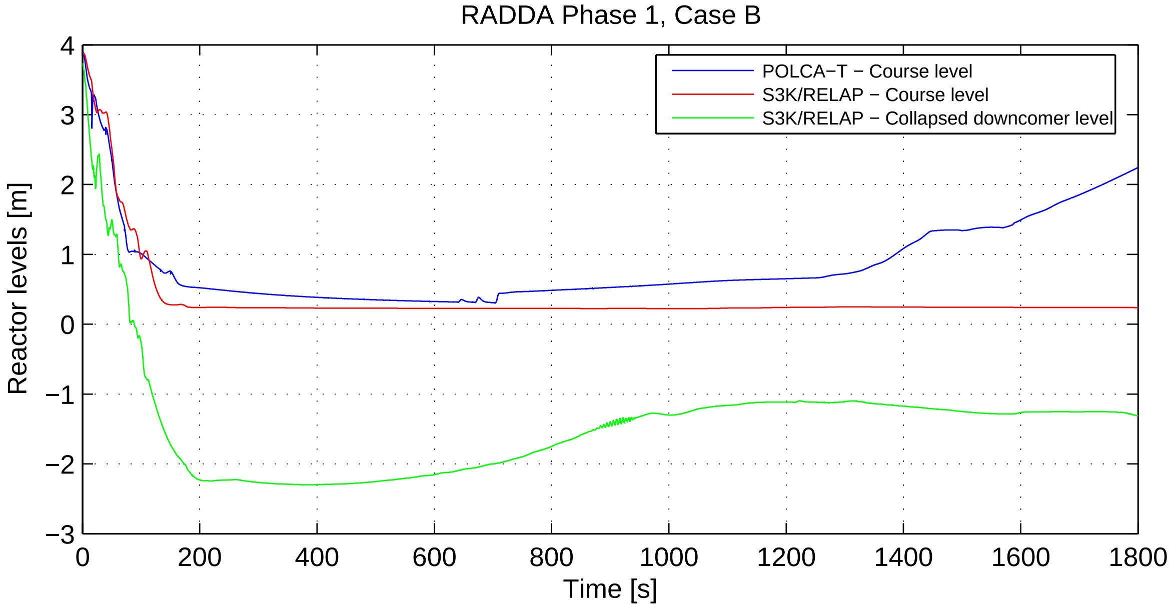

Fig. 2.5. Reactor water level according to the coarse range measurement and the collapsed downcomer water level from S3K/RELAP5 in Case B, loss of feedwater with 2 faulty open safety relief valves, no fast scram, screw insertion with 81 failing control rods.

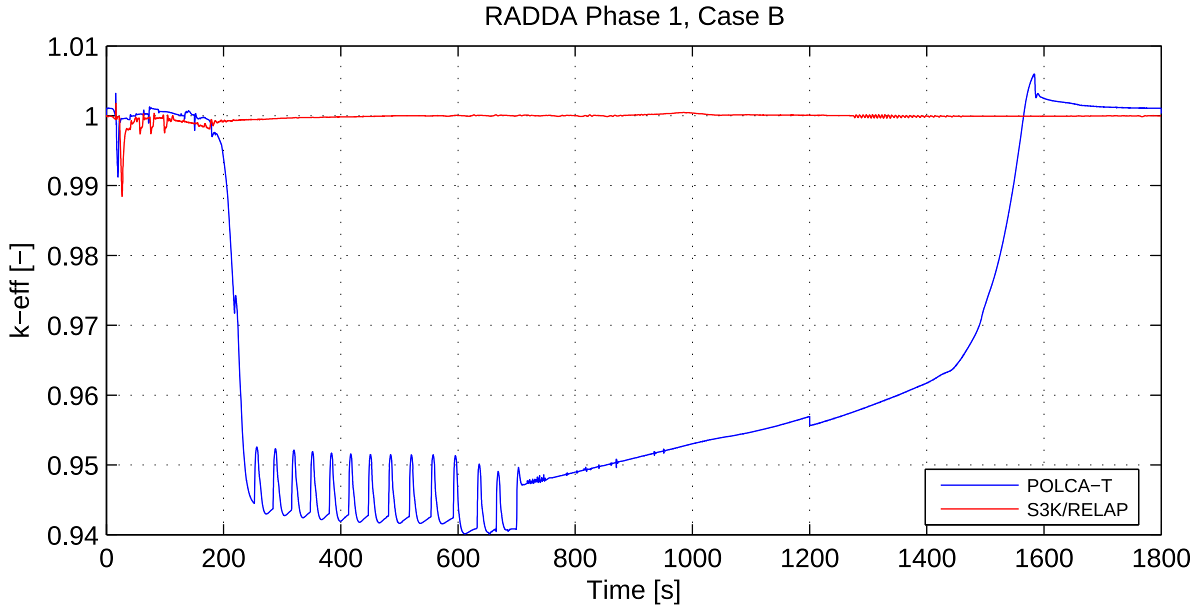

Fig. 2.6. Neutron multiplication factor as ratio of produced neutrons and absorbed + lost neutrons at actual moment of time in Case B loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

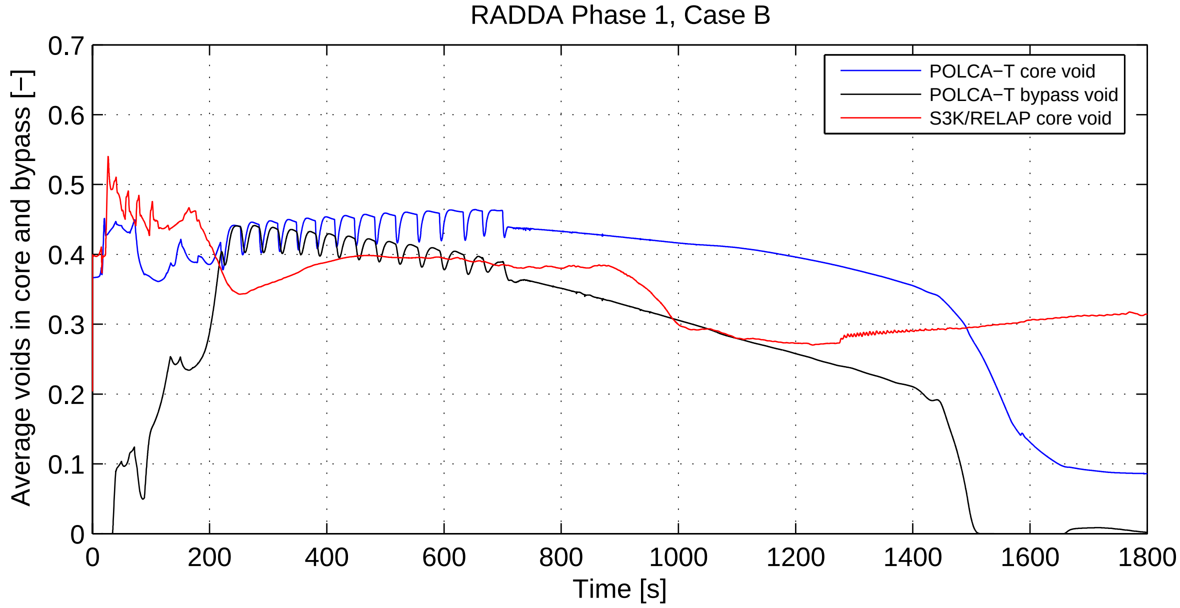

Fig. 2.7. Average void fraction of active coolant flow and of bypass flow in Case B loss of feedwater with two faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

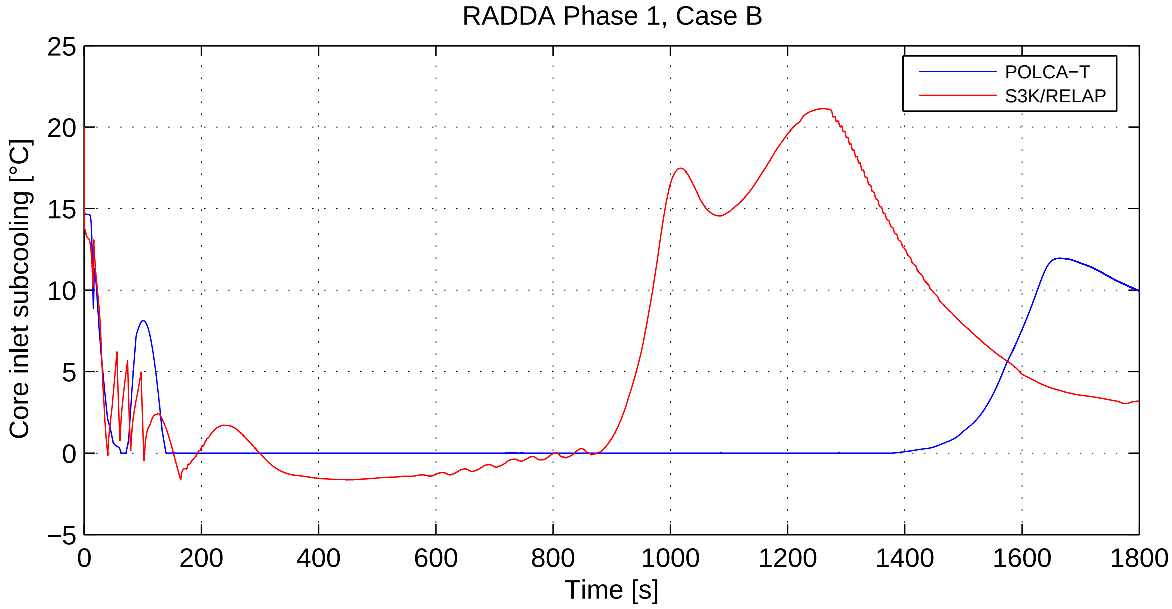

Fig. 2.8. Coolant subcooling at the core inlet in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

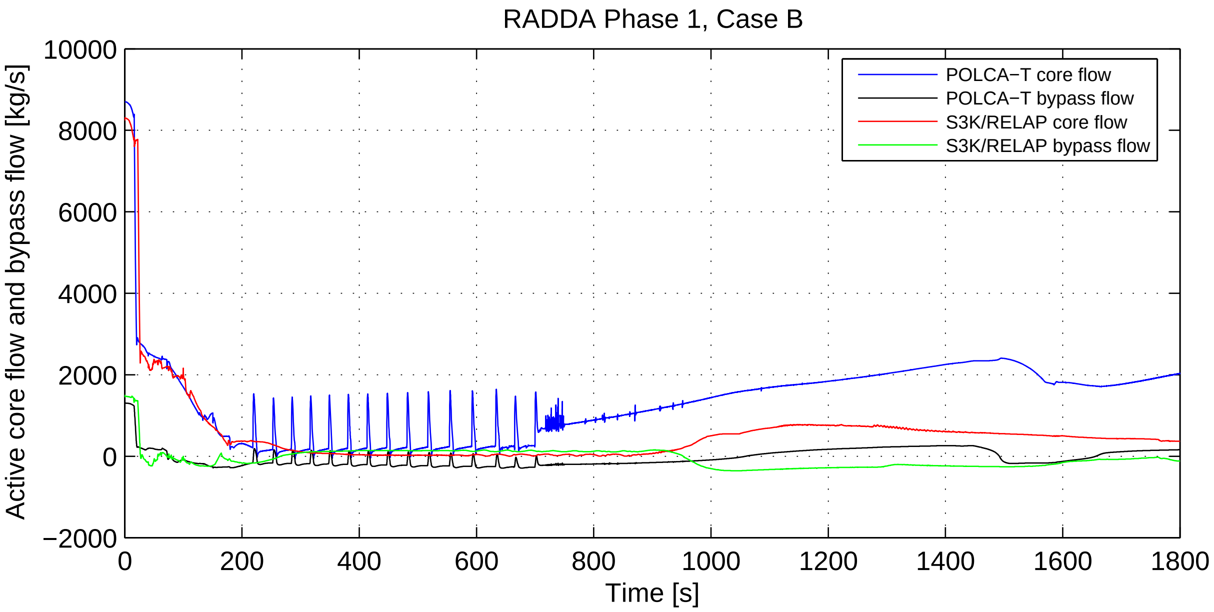

Fig. 2.9. Core coolant flow rate within fuel channels (active flow) and outside fuel channels (bypass flow) in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

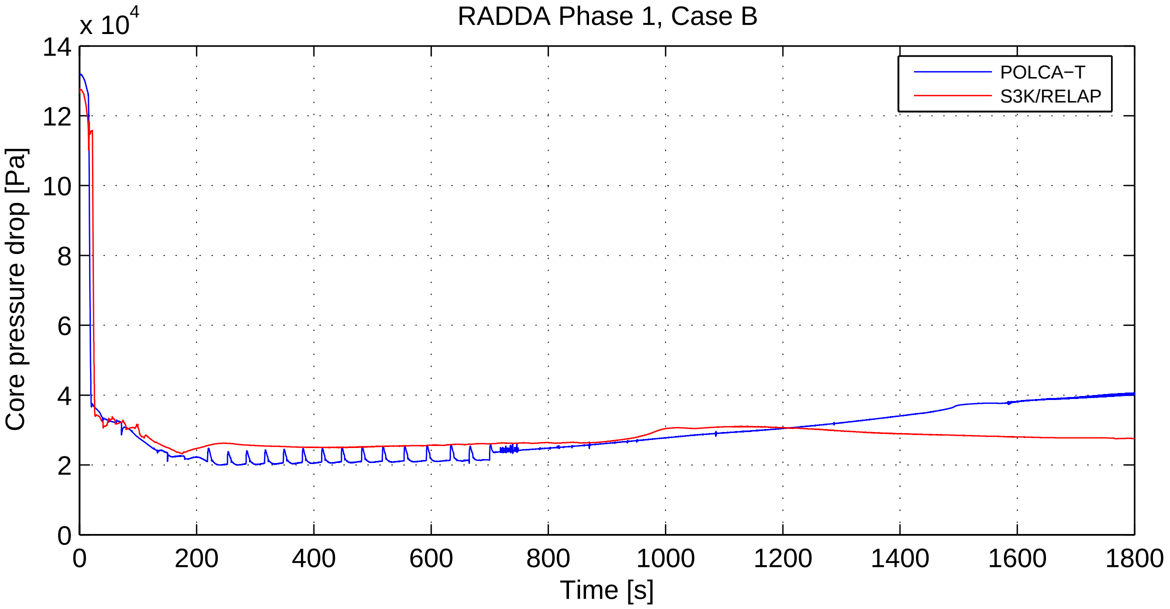

Fig. 2.10. Core pressure drop in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

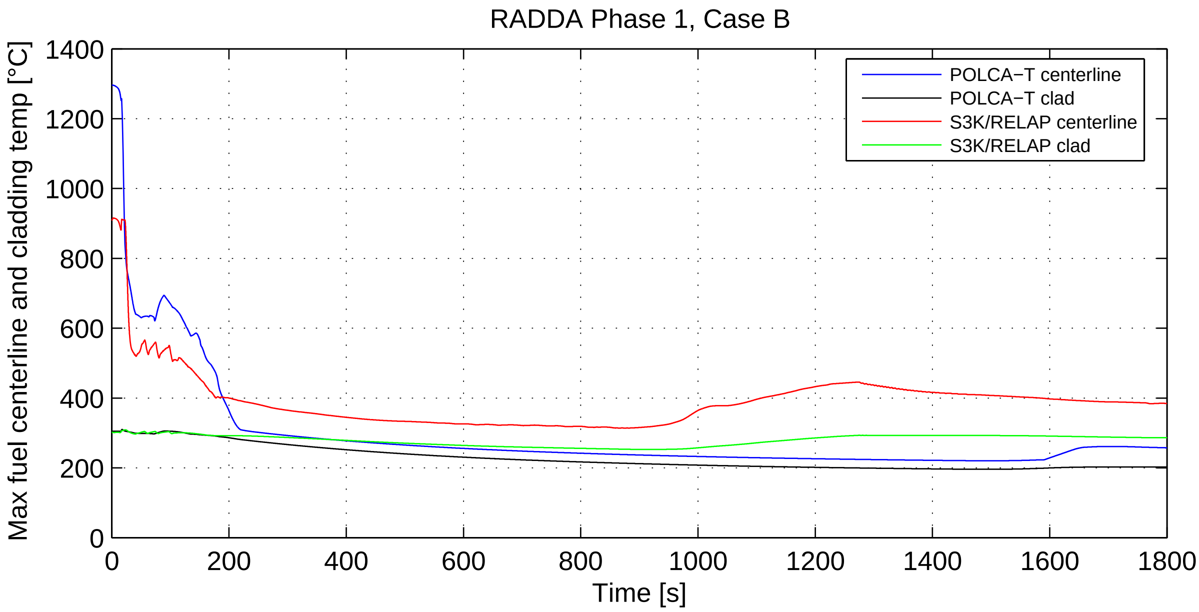

Fig. 2.11. Maximum fuel centreline and cladding temperature, for S3K/RELAP5 maximum coarse node temperature in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

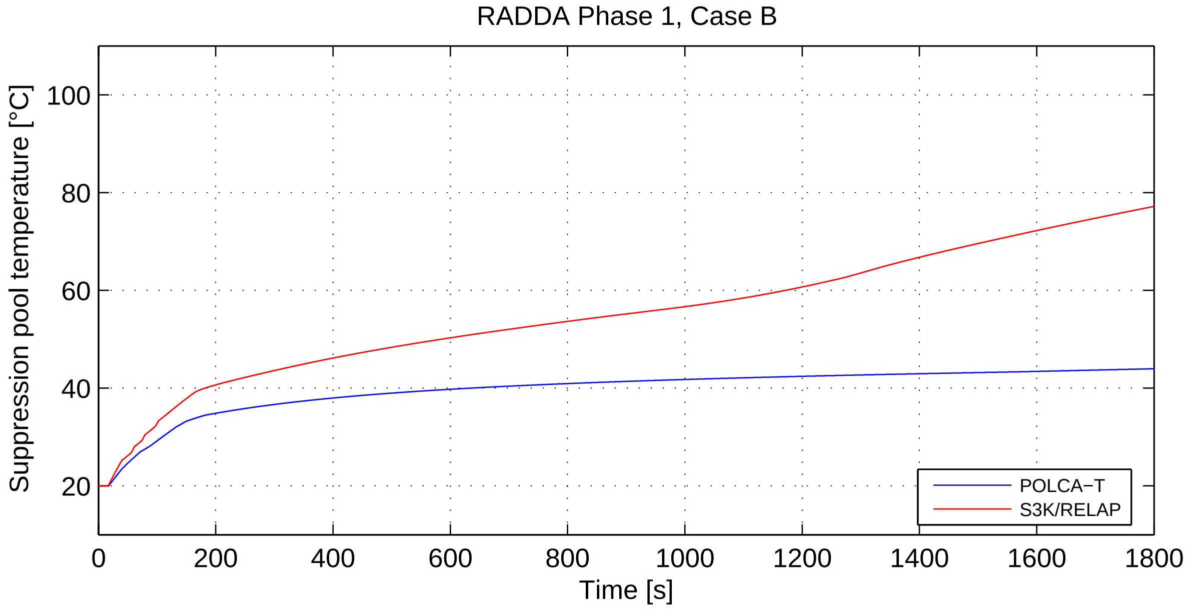

Fig. 2.12. Simple model suppression pool temperatures in Case B, loss of feedwater with 2 faulty open safety relief valves, no hydraulic scram, screw insertion with 81 failing control rods.

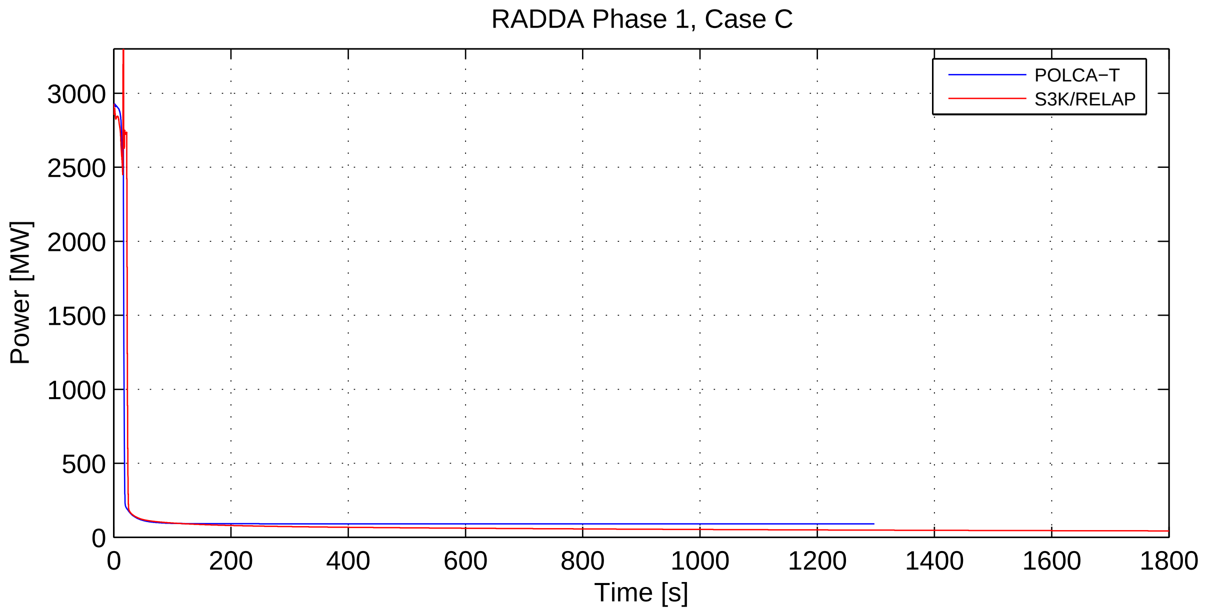

Fig. 3.1. Reactor power in Case C, loss of feedwater, 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

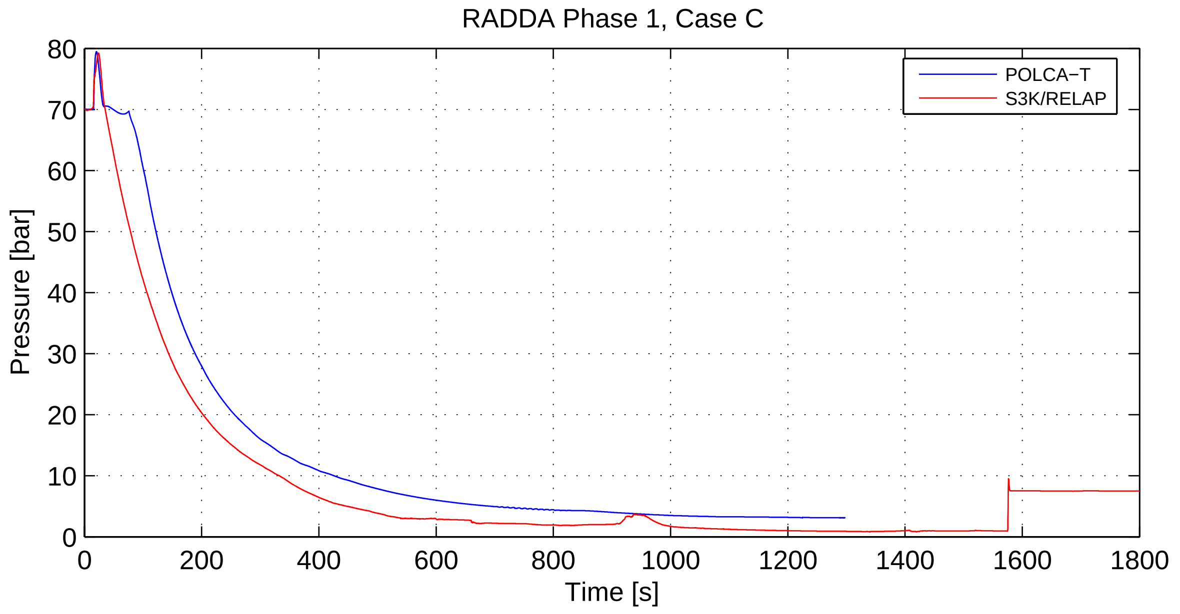

Fig. 3.2. Reactor pressure in Case C, loss of feedwater, 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

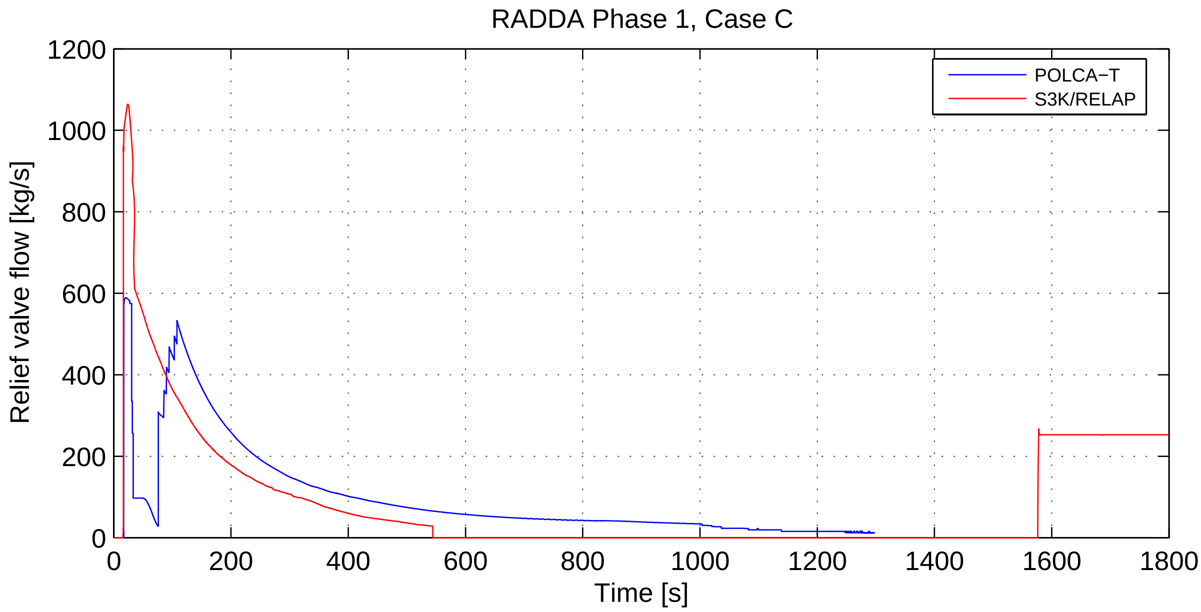

Fig. 3.3. Pressure relief valve flow rate in Case C, loss of feedwater, 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

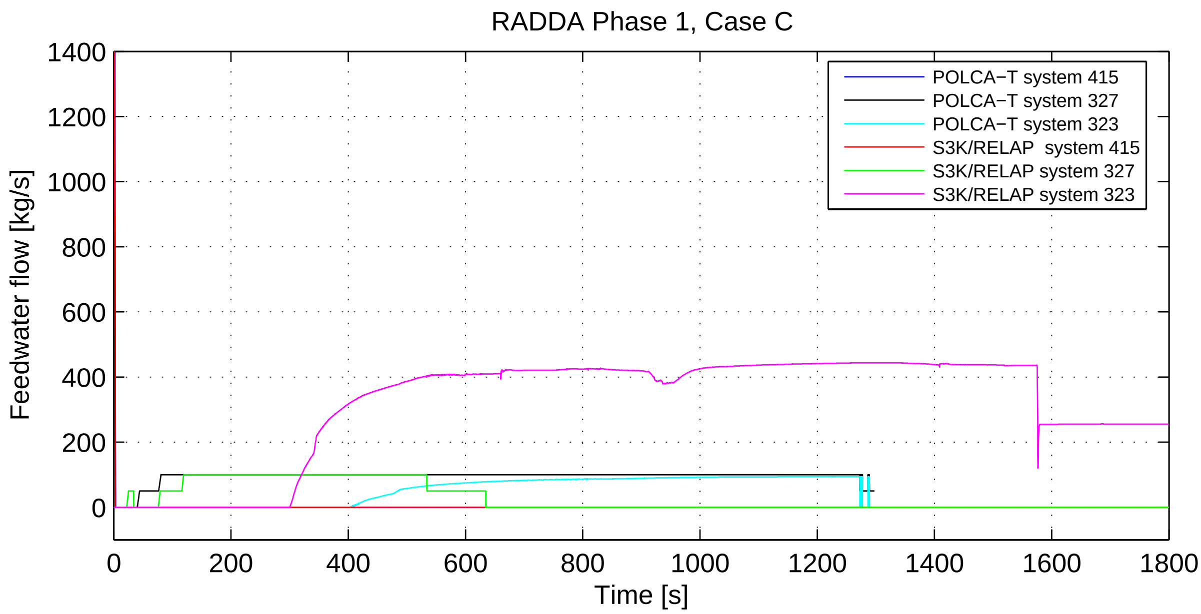

Fig. 3.4. Feedwater flow rates into the reactor pressure vessel from the feedwater system 415, the auxiliary high pressure feedwater system 327 and the low-pressure emergency core cooling system 323 in Case C, loss of feedwater, 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

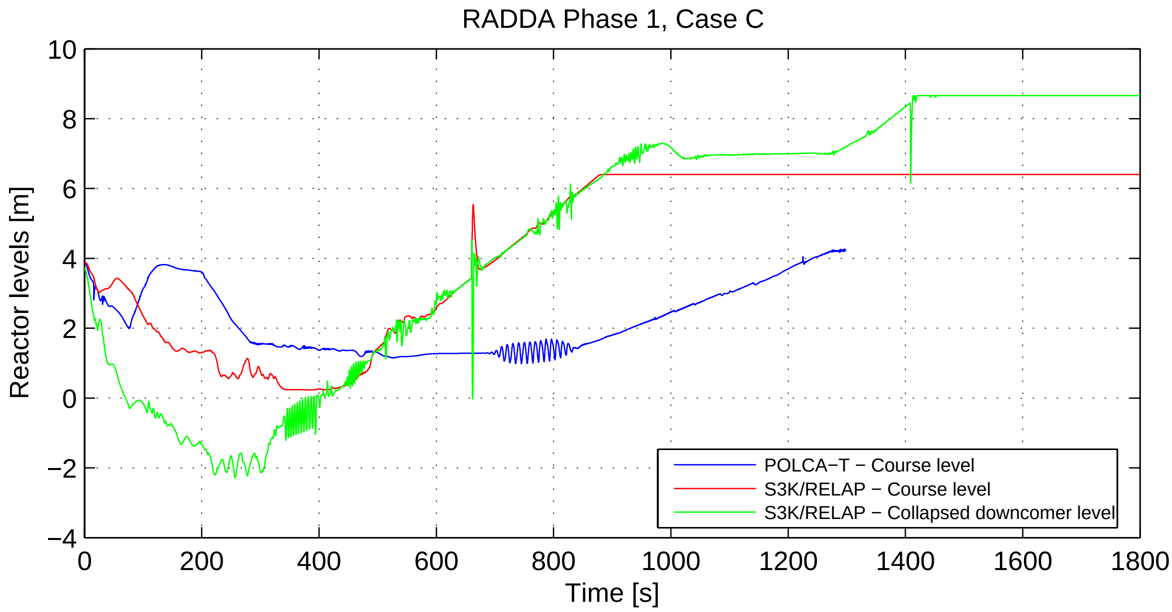

Fig. 3.5. Reactor water level according to the coarse range measurement and the collapsed downcomer water level from S3K/RELAP5 in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

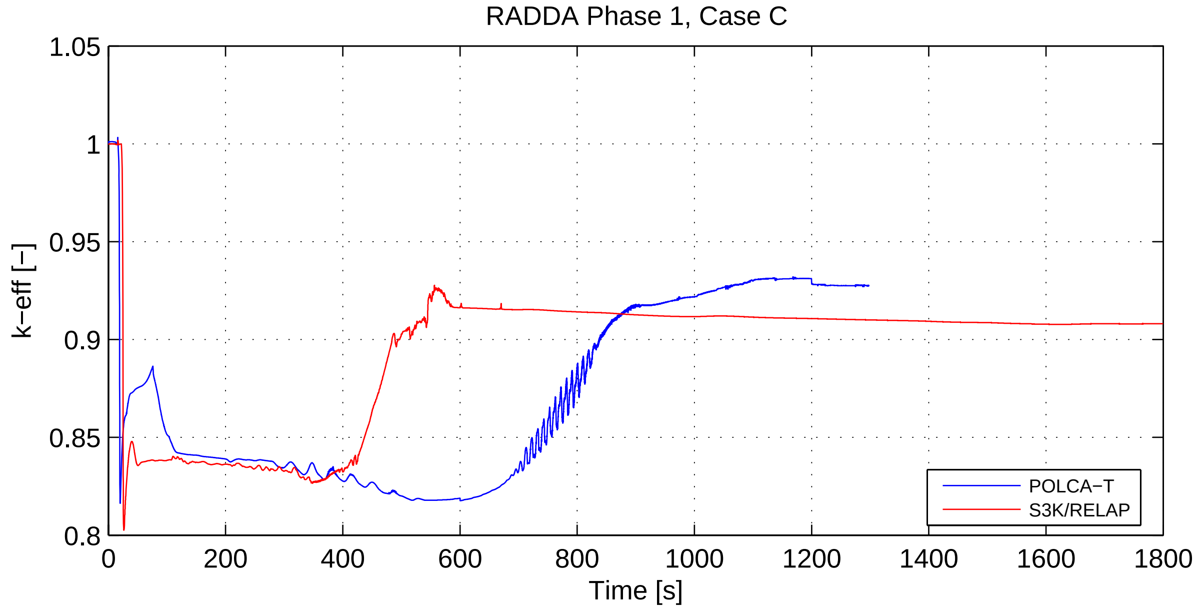

Fig. 3.6. Neutron multiplication factor as ratio of produced neutrons and absorbed + lost neutrons at actual moment of time in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram 15 failing control rods.

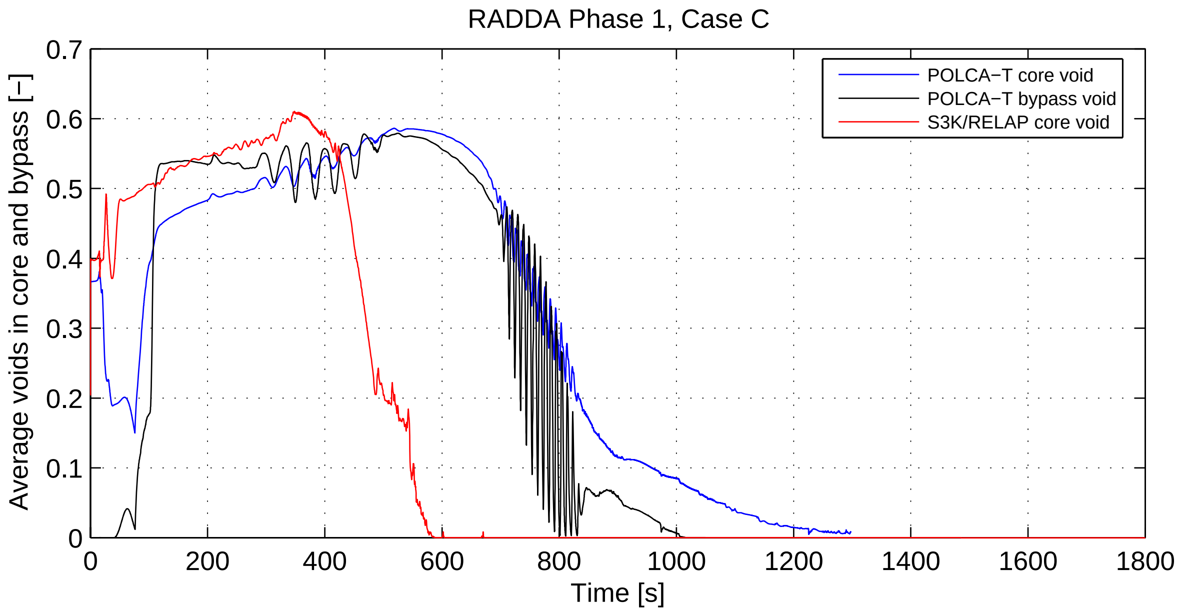

Fig. 3.7. Average void fraction of active coolant flow and of bypass flow in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram 15 failing control rods.

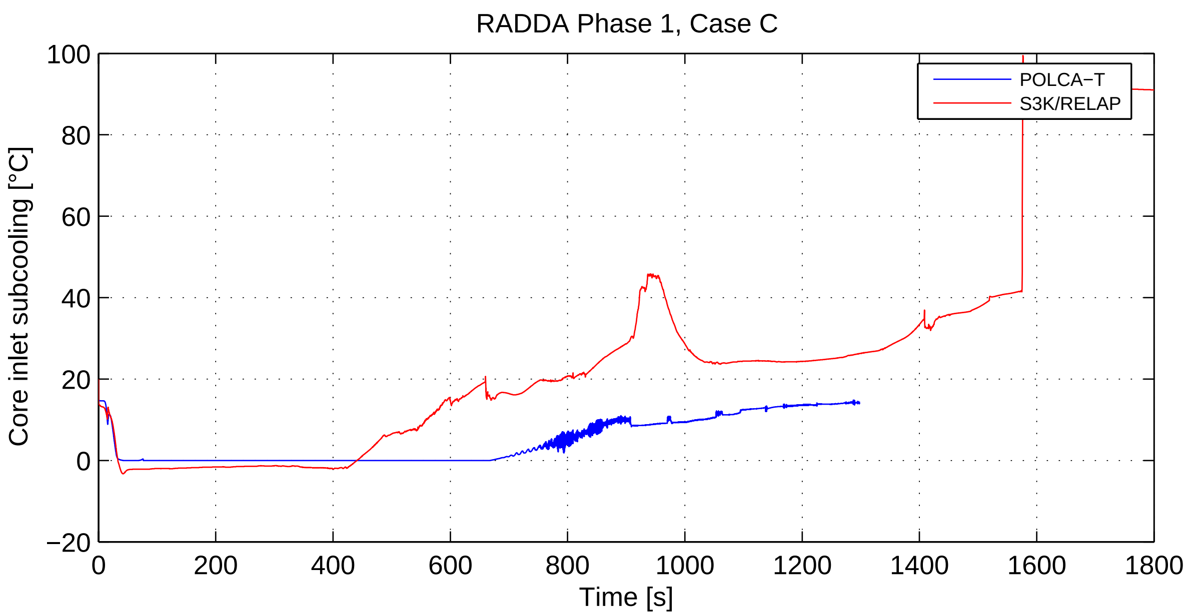

Fig. 3.8. Coolant subcooling at the core inlet in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

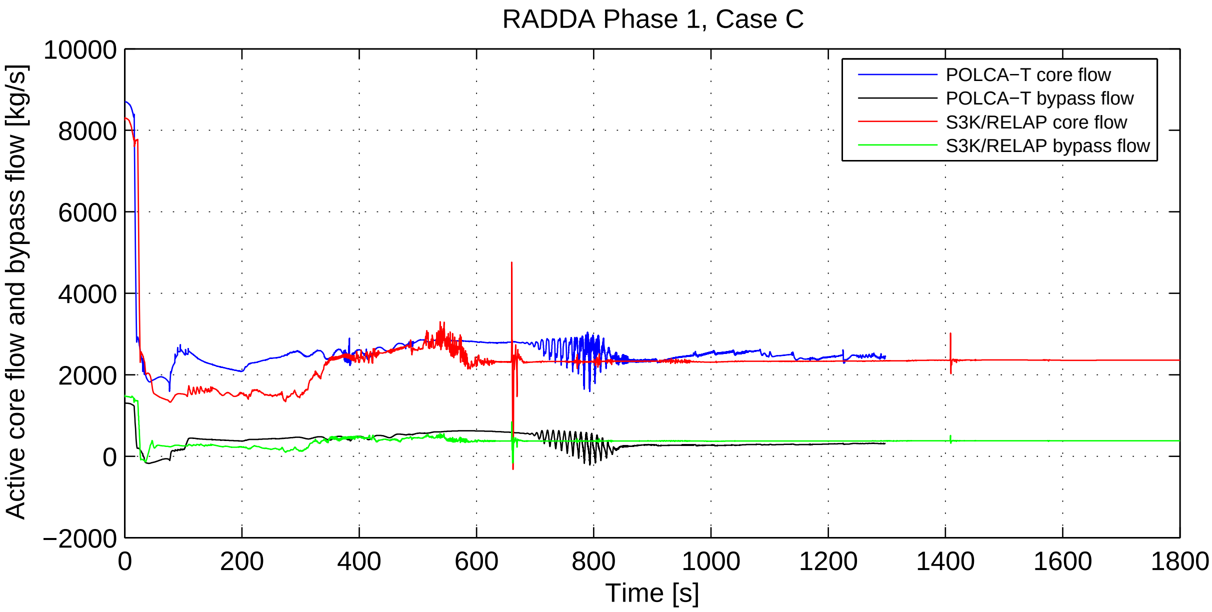

Fig. 3.9. Core coolant flow rate within fuel channels (active flow) and outside fuel channels (bypass flow) in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

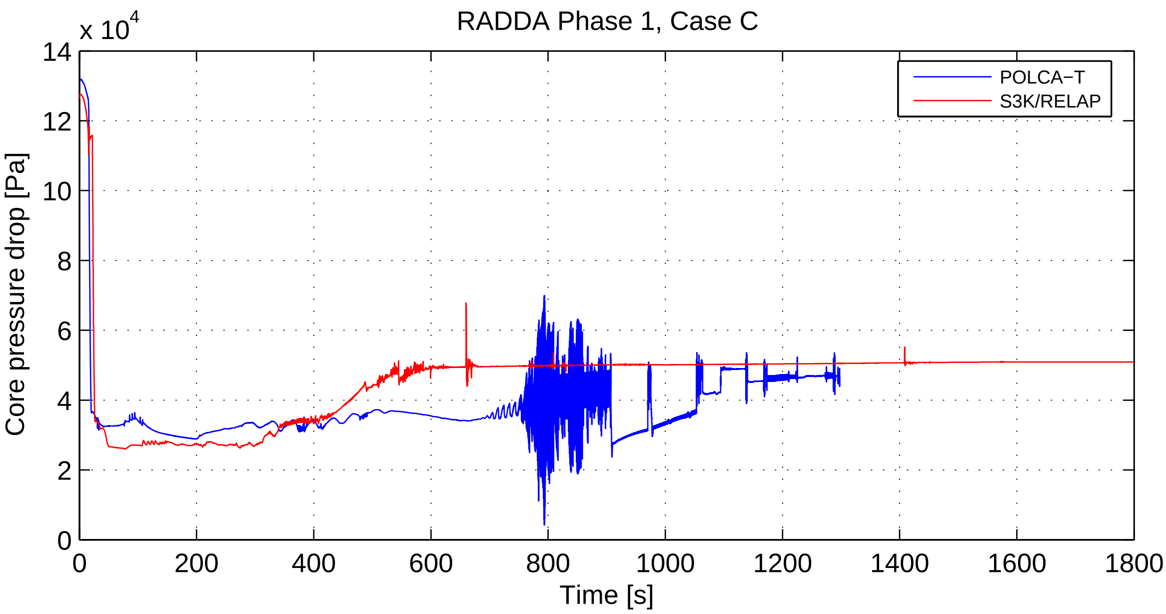

Fig. 3.10. Core pressure drop in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

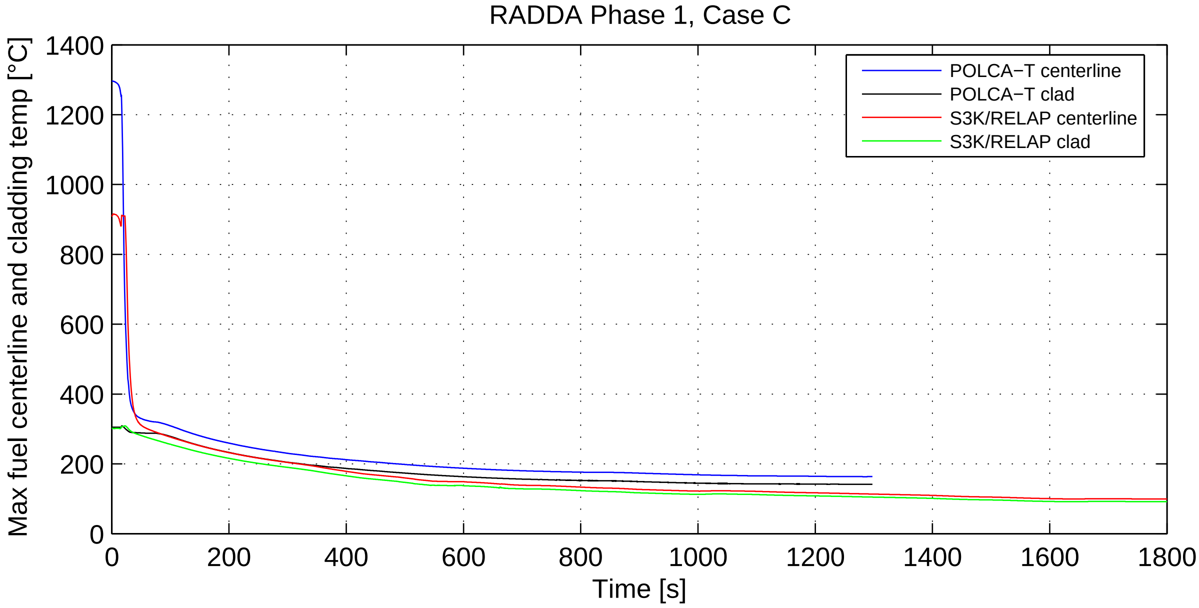

Fig. 3.11. Maximum fuel centreline and cladding temperature, for S3K/RELAP5 maximum coarse node temperature in Case C, loss of feedwater with 8 faulty open safety relief valves, hydraulic scram, 15 failing control rods.

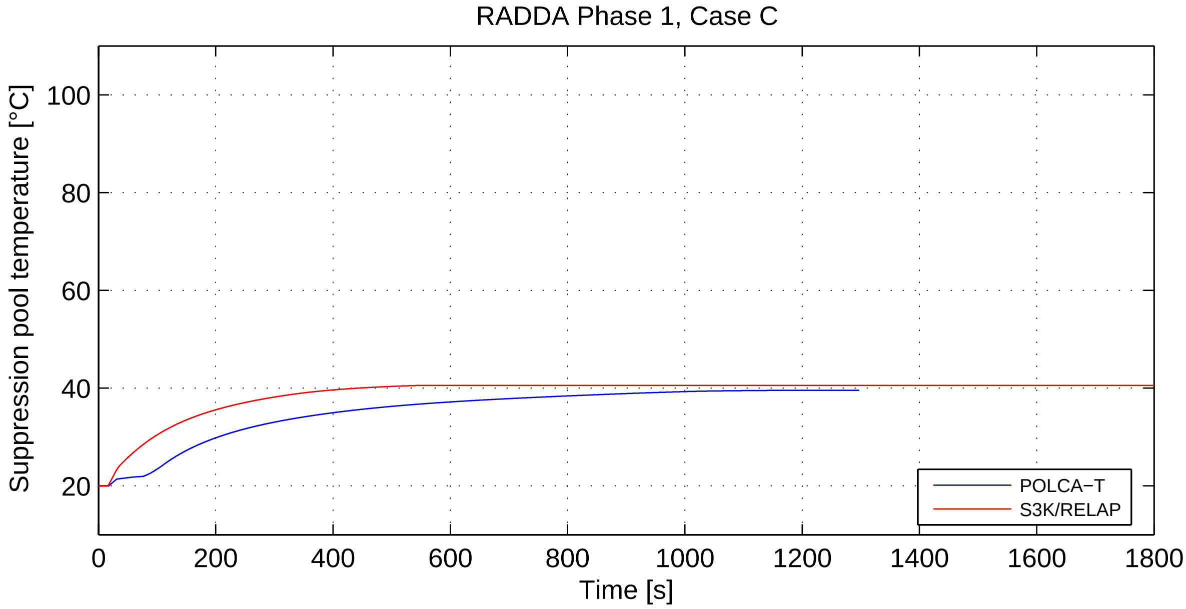

Fig. 3.12. Simple model suppression pool temperatures in Case C, loss of feedwater with 8 faulty open safety relief valves, fast hydraulic scram, 15 failing control rods.

Bibliographic Data Sheet **NKS-162**

Title	RADDA - Comparison of results of three ATWS/ATWC scenarios simulated with the help of POLCA-T and S3K/RELAP5
Author(s)	Jyrki Peltonen
Affiliation(s)	Forsmarks Kraftgrupp, Sweden
ISBN	978-87-7893-227-3
Date	March 2008
Project	NKS-R / RADDA
No. of pages	39
No. of tables	4
No. of illustrations	3 + 17
No. of references	12
Abstract	The effects of ATWS and ATWC-events with control rods failing to enter the core has been evaluated in this project. To understand the uncertainties in using modern 3D-calulation methods two different codes were used in the project. The outputs from the two code packages were compared. Within the project the used code were first evaluated against a real event, pancake core at Forsmark 3. The results give important knowledge of the core responses for such events and on how to use different code to perform such calculations. The NKS report is only one minor part of the total project. The project was sponsored by TVO, Forsmark, OKG, Ringhals, SKI besides the NKS-funding. The results could be used for PSA-studies and for deterministically safety analysis.
Key words	ATWS, ATWC, 3D-calulation comparison, uncertainties

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