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Validation of PARCS/RELAP5 coupled codes against a load rejection transient at the Ringhals-3 NPP

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ABSTRACT

This article gives an account of the development and validation of the PARCS/RELAP5 model of the Ringhals-3 unit against a load rejection transient, which occurred on November 28, 2010. The third unit of the Ringhals nuclear power plant comprises a 3-loop Westinghouse design pressurized water reactor on the Swedish West Coast.

At first, the development of the PARCS and RELAP5 models are presented. On the neutronic side, a unique cross-section interface, allowing feeding PARCS with realistic data, was developed. The dependence of the material constants on history effects, burnup, and instantaneous conditions is accounted for, and the full heterogeneity of the core is thus taken into account. The reflectors are also explicitly represented. On the thermal-hydraulic side, all the 157 fuel assemblies are modeled individually in the code input. The model is furthermore able to handle possible asymmetrical conditions of the flow velocity and temperature fields between the loops. The coupling between the two codes is touched upon, with emphasis on the mapping between the hydrodynamic/heat structures and the neutronic nodes.

Comparison between calculated and measured parameters demonstrates that the coupled model is able to correctly represent the steady-state conditions of the plant. The validation of the coupled model against measured transient plant data is then performed. It is demonstrated that the coupled model is able to catch the main features of the transient with a sufficient level of accuracy.

1. Introduction

Many power utilities worldwide have been implementing power uprates, i.e. increasing the power output of their reactors. Such uprates are an economical way of producing more electricity at a nuclear power plant (NPP) and have attracted interest due to increased electricity prices. The uprates can be achieved by increasing the thermal power in the reactor and/or improving the thermal conversion efficiency in the secondary side, e.g. by replacing the turbines and feedwater heaters (International Atomic Energy Agency: IAEA, 2011).

The Swedish Nuclear Safety Authority/SSM (then The Swedish Nuclear Power Inspectorate/SKI) received applications for new power uprates in Sweden, and among others, an application related to the Ringhals-3 pressurized water reactor (PWR) in March 2004.

The increase of the thermal power in Ringhals-3, which was performed via core optimization and/or use of fuel assemblies with a higher enrichment in U-235 and/or a higher density of the uranium dioxide, is carried out in two steps (Garis, 2004). During the first step, the thermal power was increased from 2778 MWth (original power level – 100%) to 3000 MWth (108%).

This first power uprate was done in 2007 without major modifications of the actual design of the plant, since new steam generators were installed in 1995. These new steam generators have a heat transfer area increased by 60% compared to the original model, which thus allow extracting more heat from the core. For the second step of the power uprate which was carried out in 2009, the thermal power was increased from 3000 MWth (108%) to 3144 MWth (113.2%). The original plan was to uprate to 3160 MWth (113.5%), however due to the delay in the licensing of the improved measurement system of the feedwater flow, the reactor was thus running at a lower power level (SSM, 2009). This second step required rather extensive analyses and major hardware modifications, mostly on the turbine side and modernization of control and protection systems.

It is essential to identify the main consequences resulting from the increased power level of the reactor, and their impact on the safety of the plant. SSM has to decide whether such power uprates still fulfill the requirements for a safe operation of the Ringhals-3 PWR. Performing an independent safety analysis of the Ringhals-3 power uprate was thus given by SSM as a research project to the Division of Nuclear Engineering, Chalmers University of Technology. Chalmers should be prepared to analyze unintentional control

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rod bank withdrawal at power, feedwater disturbances, Loss-Of-Coolant Accidents (LOCAs), steam line breaks, local boron dilutions and load rejection. Based on the preliminary analyses of those, a few limiting transients were selected together with SSM.

The first step of this project was the development of a complete model of the Ringhals-3 PWR. A nuclear power plant is a strongly coupled and complex system, in which there is a strong interaction between the neutron kinetics and the thermal-hydraulics. Consequently, only coupled neutronic/thermal-hydraulic tools are able to solve this kind of problem.

During the last decade, the use of coupled neutronics and thermal-hydraulics code for safety analyses has been growing intensively. IAEA (2003) organized a special meeting regarding the use of the coupled codes and concluded that it represented a very useful tool for safety analyses of the current and new reactor designs, as well as a better assessment of uncertainties associated with the use of best estimate codes. The coupled code is also efficient when it is applied to calculate the complex 3D phenomena in core. Validation of the coupled codes against some benchmark exercises has been performed in the frame of international cooperation (Ivanov et al., 2007). The use of particular codes such as Purdue Advanced Reactor Core Simulator (PARCS) and Reactor Excursion and Leak Analysis Program (RELAP5) has been reported. For example, Bousbia-Salah et al. (2006) and Espinoza et al. (2006) investigated the coolant transient in water cooled water moderated energetic reactor (VVER-1000). The turbine trip transient and flow stability test of boiling water reactor (BWR) have been reported as well (Bousbia-Salah et al., 2004; Bousbia-Salah and D'Auria, 2006), while D'Auria et al. (2003) reported their analysis on a main steam line break in Three Miles Island. The analysis of PWR using the coupled PARCS/RELAP5 has also been published quite recently (Barrachina et al., 2011; Miró et al., 2011).

In the current framework, a load rejection transient was chosen. This is an interesting but also challenging transient for coupled calculations because the power of the reactor is gradually decreased to about 60–70% for a "house-load" operation, implying a rise of important feedback effects. For our analysis, the PARCS v3.0 (Downar et al., 2010) and the RELAP5 mod 3.3 (U.S. Nuclear Regulatory Commission: U.S. NRC, 2006) codes were used for the neutron kinetics and the thermal-hydraulics, respectively.

This article describes the activity to build a coupled RELAP5/PARCS model of the 3-loop Westinghouse type PWR and the validation against a load rejection transient. The structure of this article is as follows. First, the description of the Ringhals-3 unit is presented, as well as the transient involved, followed by the development of the neutronics and the thermal-hydraulic counterpart model of the plant. Some results of the calculations are touched upon and parametric studies of important parameters are discussed. Conclusions and possible future works are mentioned in the final section.

2. Description of the plant

Ringhals nuclear power plant is situated on the west coast of Sweden, 60 km south of Gothenburg. Ringhals has four reactors: three PWRs and one BWR, and generates 28 TWh of electricity annually, supplying approximately 20% of Sweden's total electrical energy consumption.

Ringhals-3 is a pressurized water reactor which started its commercial operation in September 1981. The reactor is supplied by Westinghouse, the steam generators are from Framatome/Siemens AG, while these turbine-generators are supplied by ASEA STAL AB.

The technical data of this power plant are presented in Table 1.

Table 1
Technical data of Ringhals-3 operated at 113.5% uprated power.

Parameter	Unit	Value
Nuclear steam supply system (NSSS) power	MW	3160
Reactor power	MW	3151
Reactor coolant flow	kg/h	50.5×10^6
Reactor coolant pressure	bar	155.1
Core outlet temperature	K	598.35
Core inlet temperature	K	556.45
Steam pressure	bar	60.61
Steam flow	kg/s	1679.8
Steam temperature	K	549.35
Feedwater temperature	K	483.65

Source: (Ringhals, 2006).

3. Description of the load rejection transient

This particular transient occurred on November 28, 2010, as a scheduled, fully instrumented test. The purpose of this test is to verify the reactor transient behavior and observe the capability of the control system of preventing generator trip after the power uprating to 3144 MWt and after the modernization of the turbine control and protection system (Ringhals, 2010).

At 10:59 the generators are disconnected from the net, which means that the load of the generator decreases to only the power needed for the plant. The turbine will then increase its number of revolutions since there is no resistance. The frequency of the plant electric grid will also increase. This activates the acceleration limiter and the high pressure turbine control valves (TCV) are closed. The turbine frequency regulator will then adjust the number of revolutions to 3000 rpm again by opening the high pressure turbine control valves to 5–7% and the rest of the steam is redirected through the steam dump valves (SDV).

During the earlier stage of the transient, the increase in turbine revolution causes the reactor coolant pumps (RCP) to speed up, thus increasing the reactor coolant flow to about 104% of the nominal flow before it finally slows down again. This event consequently increases the reactor power as more moderation due to an improved reactor cooling.

When the high pressure turbine control valves close, the pressure in the steam generators will increase and consequently also the temperature, leading to an increase in the primary system temperature. The rise in the primary system temperature makes the water expands, thus increasing the pressure and level of the pressurizer.

When the turbine pressure drops, the control rods will be inserted with maximum speed, thus decreasing the reactor power. As a consequence of the rapid insertion of control rods, the axial power distribution is then shifted outside the operation band for delta flux. In order to get the power distribution within range, the operators start boration which in those cases are done via a high concentration boric acid tank (BAT). At this stage the main objective for the operator is to restore the axial flux within the accepted range (Svensson, 2010). After 2 min the control rods are manually maneuvered which stop further insertion of the rods and also power reduction. The reactor power is then almost constant at 70%.

4. Methodology for the coupled model

In this section, the methodology for creating a coupled model of the Ringhals-3 PWR is presented. The main purpose of the neutronic modeling is the development of a PARCS model corresponding to the Ringhals-3 unit, so that such a model can be later on coupled to the RELAP5 thermal-hydraulic code. Since history effects are important to be taken into account while modeling a nuclear core, the complete history of the Ringhals-3 reactor had to be recovered. The SIMULATE-3 code (Covington et al., 1995) was used to model the Ringhals-3 reactor from the start of its operation, since Ringhals AB, i.e. the power utility operating the four Ringhals units, has been using SIMULATE-3 for in-core fuel management and core follow of Ringhals-3. History variables such as the history of the moderator density and of the boron concentration were then transferred from SIMULATE-3 to PARCS. Such a transfer was made possible since a unique cross-section interface was developed for feeding PARCS with realistic sets of material constants for any heterogeneous light water reactor (LWR) core, i.e. the dependence of the data on instantaneous variables, history variables and burnup is fully accounted for.

The current RELAP5 model of Ringhals-3 was built on the basis of a legacy input created by Studsvik EcoSafe (Eriksson, 1994) for an earlier version of RELAP5 in 1994. Obviously, the RELAP5 code itself has gone through extensive changes in over a decade period. Therefore, the model had to be thoroughly modified in order to satisfy the syntax rules of the new code version; to represent the current situation at Ringhals; but most crucially for having a very detailed core description in accordance with the PARCS model.

4.1. CASMO/SIMULATE

All the SIMULATE-3 input files modeling the complete history of the plant since the start of its commercial operation until December 2010 were obtained from the Reactor Physics group at Ringhals AB, representing a total of 28 fuel cycles, upon which the first 27 are full cycles and the last cycle is partial. The binary library file containing the results of the transport calculations for each of the fuel/reflector segments loaded in Ringhals-3 was also obtained. However, Ringhals AB loaded a new type of fuel assemblies, so called shielding assemblies/SA in 2009 (Sandberg et al., 2010) and the corresponding binary file of this segment was not fully compatible with the interface code to create the PMAXS files. New cross-section sets for the SA were thus created using CASMO-4 (Ekberg et al., 1995) and the resulting new binary file was used instead for subsequent calculations.

The SIMULATE-3 simulations thus provide the full three-dimensional spatial distribution of the history of the moderator density, of the history of the boron concentration, and of the burnup throughout the core. The knowledge of these parameters allows retrieving the correct material data for each fuel/reflector segment in the PARCS model, provided that the instantaneous conditions (control rod insertion, moderator density, boron concentration, and fuel temperature) can also be determined.

4.2. Interface code

To obtain macroscopic cross-sections, discontinuity factors, kinetic data, microscopic cross-sections and yields for the poisons, PMAXS files were used. These files were constructed from data obtained from CASMO-4 calculations. The creation of the PMAXS files was done with an in-house interface code, which is based on the use of SIMULATE-3 to read the CASMO-4 library file. To use this interface one needs the CASMO-4 library file and SIMULATE-3 restart file for the core to be studied. The first step in the procedure is to obtain information on which fuel types are loaded in the core. When this is known, an edit of all relevant parameters for all fuel types at all conditions found in the library file is requested. Finally these data are converted to the PMAXS formalism and written to the PMAXS files. There is only one set of kinetic data for the whole core, this means that there is no spatial dependence for the kinetics data. More information on this cross section interface can be found in Stålek and Demazière (2008).

4.3. PARCS

The first step was to model in PARCS an actual core loading. In this article, only the core loading corresponding to the fuel cycle 28 is considered. One hundred and fifty-seven fuel assemblies constitute the active core. An explicit treatment of the bottom, top, and radial reflectors was carried out. Radially, the fuel region is made of a combination of 11 different fuel assemblies, while the reflector region is constituted by 2 different segments (one segment only taking the baffle into account, and another segment taking both the baffle and the core barrel into account). Axially, each fuel/reflector assembly in PARCS is modeled by an active region corresponding to the core active height, surrounded by a bottom reflector and a top reflector. Some fuel assemblies even present an axial zoning of the core active height. Combining the radial and axial zoning of the core, as well as the different types of fuel and reflector assemblies, 15 different segments are necessary to completely define the core. The second step in the setup of the PARCS model was to associate a PMAXS file to each of these 15 segments. Although only 15 PMAXS files are necessary to fully describe the core, the actual heterogeneity of the core was accounted for, i.e. each node is represented by a set of variables (burnup, history variables, and instantaneous variables) which differ significantly throughout the core. The core is actually represented by 5,746 regions [i.e. 26 axial nodes $\times$ 221 radial nodes (157 fuel assemblies and 64 reflector assemblies)]. Thereafter, the spatial distribution of the exposure and of the history variables (moderator density and boron concentration) throughout the core were obtained from SIMULATE-3 at a cycle exposure of 1.592 GWd/tHM and were fed into PARCS.

The results of a stand-alone PARCS calculation are represented in Figs. 1 and 2. As can be noticed in Fig. 1, the axial power distribution calculated by PARCS has a very good agreement with the measured data, having a root mean square of errors of 0.0475. Most of the fuel assemblies have less than 10% of relative error in the radial power distribution with an overall root mean square of errors of 0.0284, whereas larger relative errors can be found in the periphery which has low power ratio anyway (see Fig. 2(b)).

4.4. RELAP5

The development of the model was carried out in successive steps. At first, the original structure of the cold leg (CL), including the ECC mixer components were modified, following the explicit recommendations of the code User's Guide and Input Requirements. Further structural changes were necessary on the secondary side. In order to capture the pressure pulse propagation, the number of sub-volumes of the steam line was increased by a factor of 10.

The largest modifications were related to the reactor pressure vessel internals. The initial model had a very coarse nodalization, including only one downcomer and two channels in the core: one "hot channel" representing the hottest fuel bundle, and one "average channel" representing all the rest of the core.

Therefore, the most significant refinement was done in the core region by modeling each of the 157 fuel assemblies individually, both for the hydrodynamics and for the heat structures. The numbering scheme of the individual fuel assemblies is shown in Fig. 3.

At the initial phase of the model development, the frozen version of RELAP5, which was available for the users, was Mod3.3gl Patch 03. This particular version was not able to handle a full radial and axial core model (157 radial channels and 24 axial nodes at the same time), due to the limited size of memory container built into the code. Thus, a compromise had to be made with priority on the higher radial resolution. It was then decided to use 8 axial levels, while maintaining the 157 channels for one-by-one radial

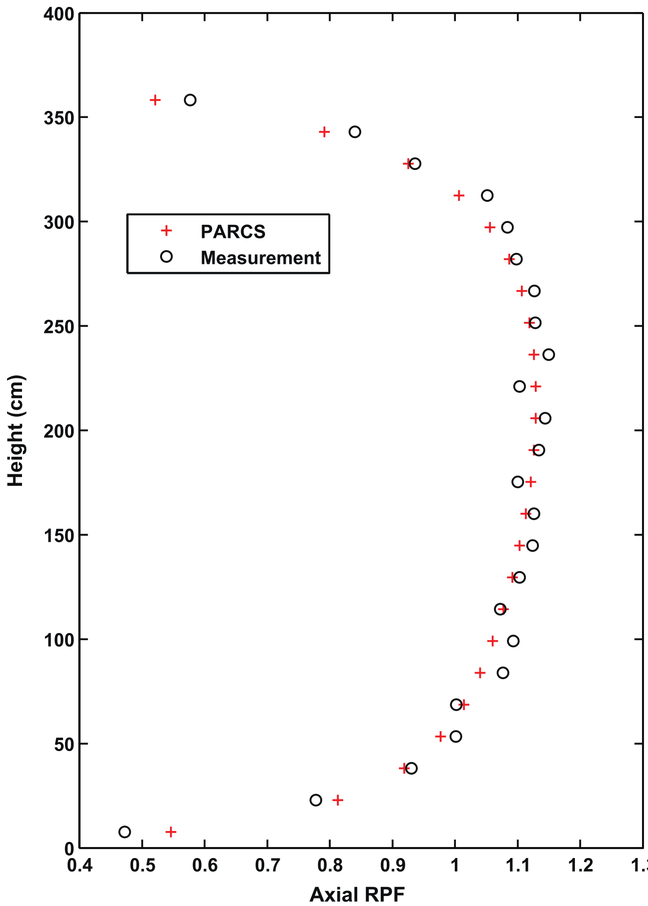
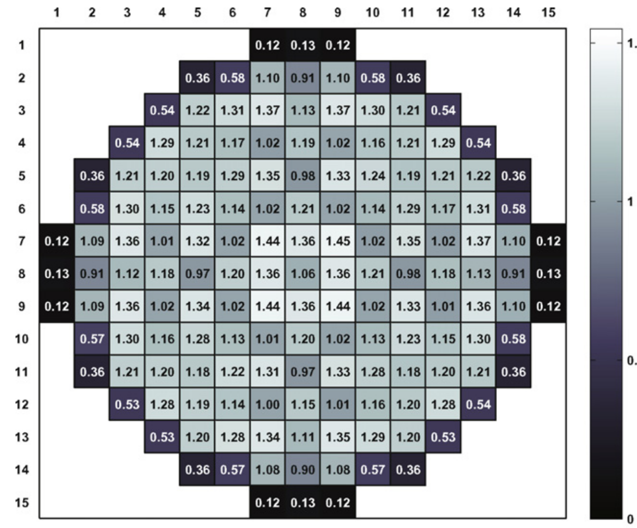
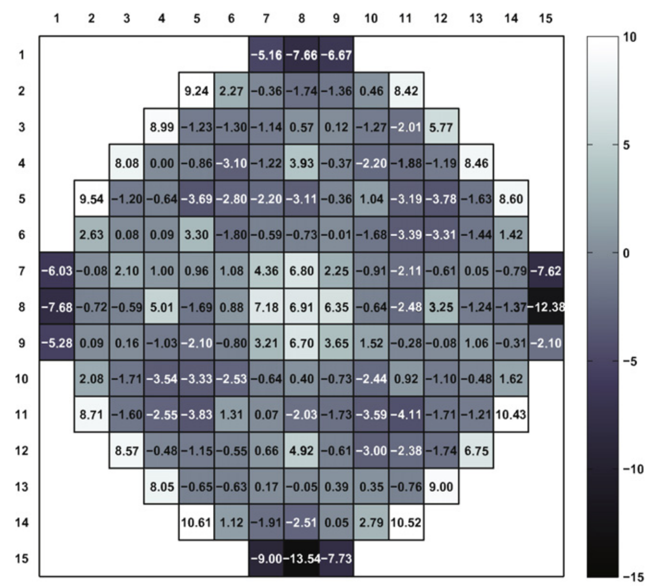


Fig. 1. Comparison of the axial relative power fraction as calculated by a stand-alone PARCS and measured data at 1.592 GWd/tHM. Root mean square of errors = 0.0475.



(a) Radial relative power distribution



(b) Relative errors in %

Fig. 2. (a and b) Comparison of the radial relative power fraction as calculated by a stand-alone PARCS and measured data. Root mean square of errors = 0.0284.

mapping with the neutronic model. An extension of the model to 24 axial levels is in fact already in the roadmap for implementation in the future, as we now have access to the latest version of RELAP5.

The core inlet and outlet needed special considerations due to a limitation in RELAP5, that a branch component may be connected to maximally 9 other volumes. Arrangement of altogether 157 junctions to the fuel assemblies was realized by application of 3×6 additional branch components both at the bottom and the top of the core (see Fig. 4). It is essential that some specific phenomena, for instance an asymmetric behavior of the loops should be captured properly. For this reason, the downcomer and the core were split into 3 parallel channels in order to retain the 3-loop structure of the primary side even within the reactor pressure vessel (RPV). It is also important to distinguish between the coolant flowing through the fuel assemblies while being heated, and the remaining part of the main loop flow. Thus, the core has been extended with altogether 3 bypass channels per loop: one is representing the baffle–barrel space, the other one is modeling the open guide thimbles, and the third channel is created for the flow path at the core periphery.

Beyond the RPV, the current nodalization of the Ringhals-3 model includes the following major parts: (i) 3 hot-legs and 3 cold-legs, (ii) a pressurizer model with spray system and electrical heaters, (iii) 3 main circulating pumps, (iv) residual heat removal systems, (v) 3 steam generators with vertical heat exchanger tubes, (vi) feedwater, charging, and letdown systems, (vii) 2 simplified turbine models, (viii) steam dumping lines, and (ix) safety and relief valves (see Fig. 5).

The control and protection systems were designed carefully and the parameters strictly follow the values laid out in the Ringhals-3 PLS document (Ringhals, 2006, 2007). The level in the steam generator (SG) is maintained by the level control system. The input signals (measured feed water (FW) flowrate, steam flowrate, narrow-range level, and the nuclear power) are basically fed into the proportional and integral (PI) type controller. The output signal determines the stem position of FW control valve in the model.

The pressurizer (PRZ) pressure and level are controlled by their corresponding control components. This complex system is modeled in a realistic way with using the same time constants and control parameters as in the plant. In case of excess primary pressure the spray system is actuated, while in case of lower pressure the proportional and on/off heaters are activated accordingly.

More discussions on the RELAP5 model and its validation for a stand-alone calculation can be found in a NUREG report (Bánáti et al., 2010).

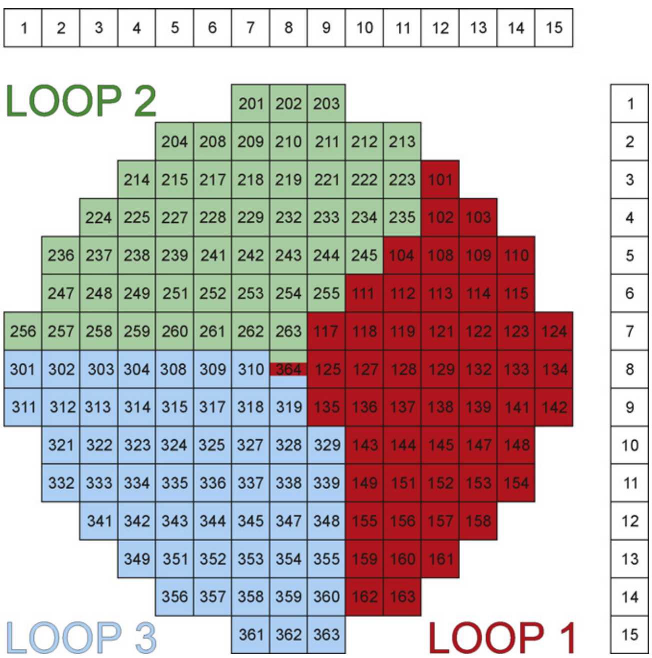


Fig. 3. Numbering scheme of the fuel channels in radial sectors.

4.5. Coupling method

The coupling between RELAP5 and PARCS is performed by means of a general interface (GI) via the message-passing protocol in the parallel virtual machine (PVM) environment, which is devoted to map transient evolution relevant data and to manage the exchange of all thermal-hydraulic and neutronic data between the two codes. The spatial coupling of RELAP5 and PARCS makes use of an internal integration scheme in which the solution of both the system and core thermal-hydraulics is performed by RELAP5, and the spatial kinetics solution is performed by PARCS. The temporal coupling of RELAP5 and PARCS is explicit and the respective field equations of the two codes are solved with the same frequency. More detail about the implementation of the spatial and temporal coupling method can be found in, e.g. Barber et al. (1998) and Kozlowski et al. (2004).

4.6. Mapping between the neutronics and the thermal-hydraulics

One of the most crucial steps in the process of preparing input files for coupled calculations is the mapping between the neutronic and the thermal-hydraulic codes. More precisely, two mappings are necessary to be set up: one mapping for the neutronic/hydrodynamic structures, and one mapping for the neutronic/heat structures. The first mapping allows specifying in the neutronic code which fuel temperature, moderator temperature/density from the thermal-hydraulic code needs to be associated with a specific neutronic node. The second mapping allows specifying in the thermal-hydraulic code which fission power needs to be associated with a specific thermal-hydraulic node. In both mappings, weighting factors have to be defined for each of the links between the neutronic and the thermal-hydraulic nodes.

Although PARCS has the ability to automatically generate the mapping file based on a minimal amount of mapping information provided by the user, the applicability of such an automatic mapping is restricted to very simple cases. For instance, the Ringhals-3 PARCS model contains 26 axial planes, 157 fuel and 64 reflector assemblies, whereas the radial nodalization is 2×2 per fuel/reflector assemblies. The number of neutronic nodes thus amounts to 22,984. This high number of neutronic nodes and the need for properly taking the effect of the reflector into account in the neutronic model prohibit the use of the automatic mapping. Tools for generating the mapping tables for the neutronic/hydrodynamic structures and neutronic/heat structures were thus created.

Weighting factors between the neutronics and thermal-hydraulics were required in each of these mapping files. A weighting factor defines the actual weight of a given cell belonging to a specific component. The sum of the weighting factors of RELAP5 components and cells belonging to a specific PARCS neutronic node should be equal to unity. The weighting factors thus represent how much a RELAP5 component and/or cell contribute(s) to a PARCS node in terms of feedback (hydrodynamic structures mapping) and how much this PARCS node contributes to the heat generated in a given RELAP5 component and/or cell (heat structures mapping). Very often, geometrical considerations can be used to estimate the weighting factors, i.e. the weighting factors can be seen as the volume fraction of each component and/or cell in the corresponding neutronic node.

Concerning the radial mapping, the RELAP5 core is modeled by 157 channels, whereas in PARCS the core is modeled by $157 \times 2 \times 2$ nodes. For the mapping of the hydrodynamic structures of the reflectors, the bottom reflector under the active core was associated to the components modeling the lower plenum, whereas the top reflector above the active core was associated to the components modeling the upper plenum. The radial reflector was associated to the RELAP5 components representing the bypass flow in the core cavity inside the core baffle, and the bypass flow in the baffle/plate gap and in the baffle/barrel region, with weighting factors of 0.272 and 0.728, respectively. These weighting factors were determined from the results of CASMO-4 calculations performed for retrieving the spatial distribution of the flux in the bypass active core/baffle, baffle, and bypass baffle/barrel regions.

Concerning the axial mapping, the active core discretization in PARCS consists of 24 axial planes of equal height, whereas the core discretization in RELAP5 consists of 8 axial planes of equal height. No axial discretization of the bottom/top reflectors was carried out, i.e. both the bottom/top reflectors and the corresponding lower/upper plenums were assumed to be represented by one axial level only.

5. Simulation of the load rejection test

5.1. Specific simulation parameters related to the transient

In order to simulate the load rejection test, the generic PARCS and RELAP5 models have to be adjusted accordingly. The movement of the control rods during the transient is modeled in the PARCS code as a boundary condition. Fig. 6 shows the comparison between the measured positions of the control rods and the input PARCS model. Note that the rod position is given in %, in which 100% represents fully withdrawn rods and 0% represents fully inserted rods. As PARCS specifies only 10 positions of the control rods as function of time in a single input file, we use several consecutive restart calculations to enable high resolution of the movement.

Two different models of the xenon and samarium poisoning are used in PARCS. During the first part of the simulation to reach a steady-state, the equilibrium poisons model is used, meanwhile during the transient part of the simulation, the transient poison model is used.

The boron concentration in the reactor coolant is set to 1165 ppm as an initial condition. This gives $k_{eff} = 1.0004542$ in the initial coupled steady-state calculation. During the transient

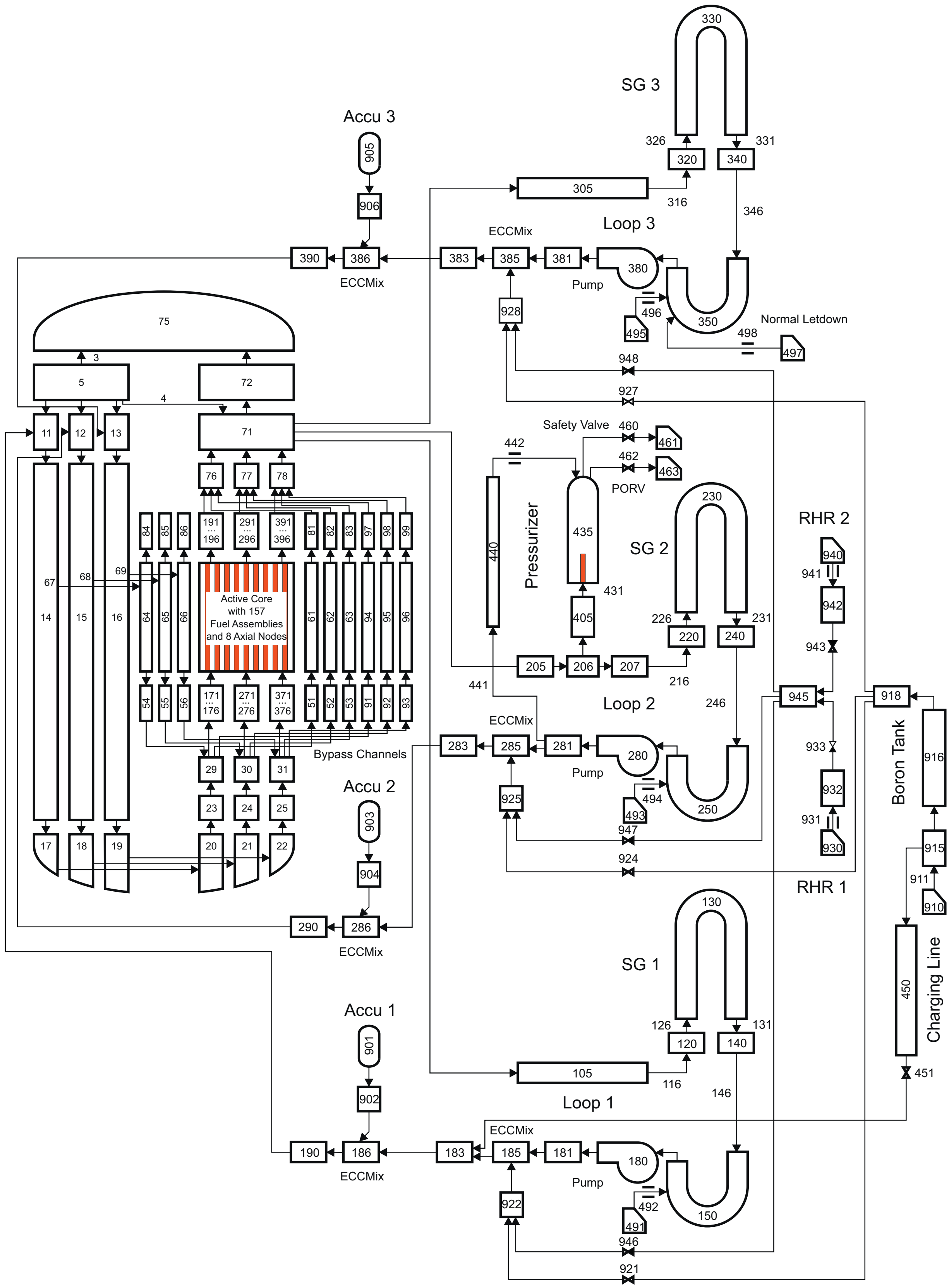


Fig. 4. RELAP5 nodalization of the primary side.

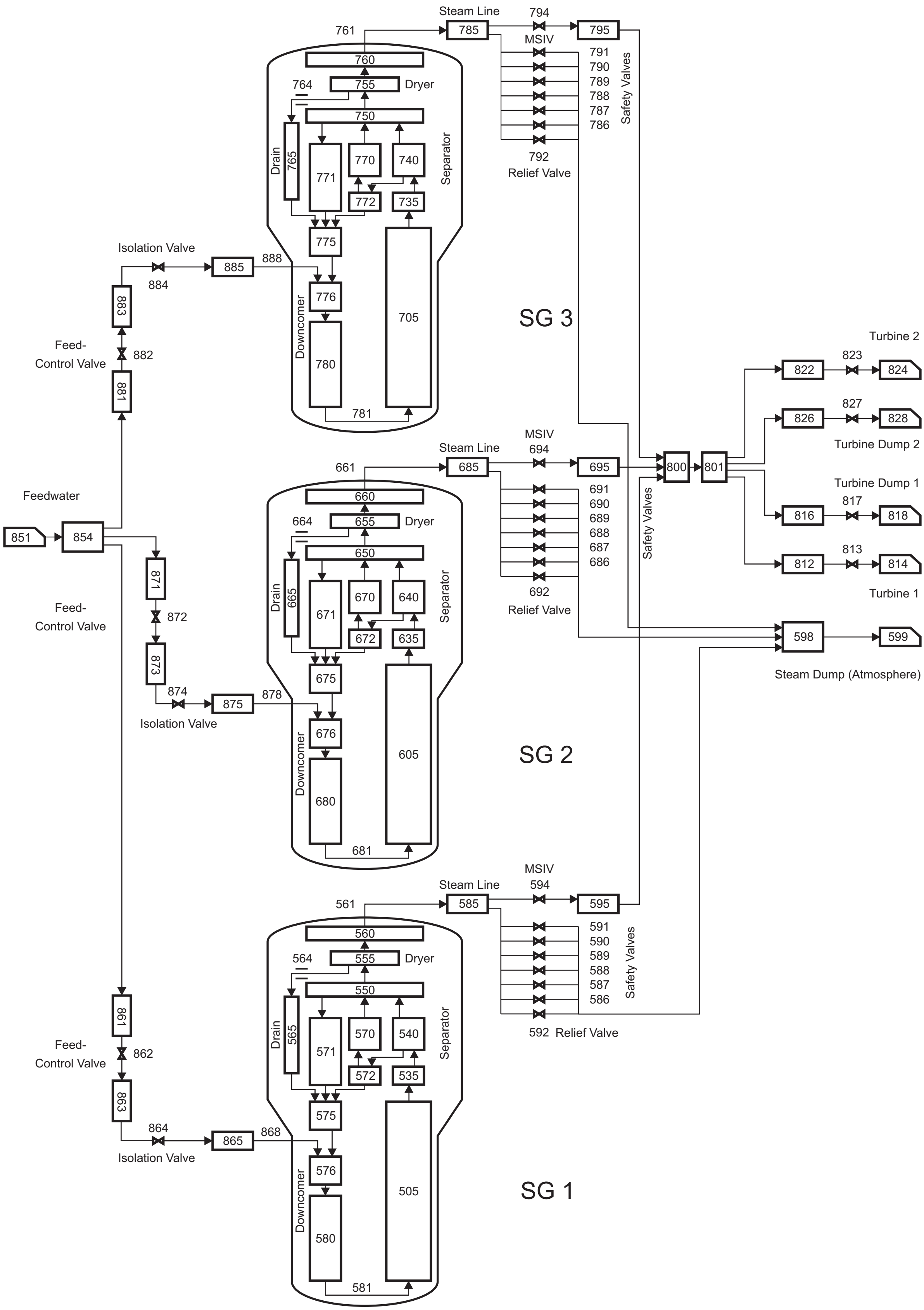


Fig. 5. RELAP5 nodalization of the secondary side.

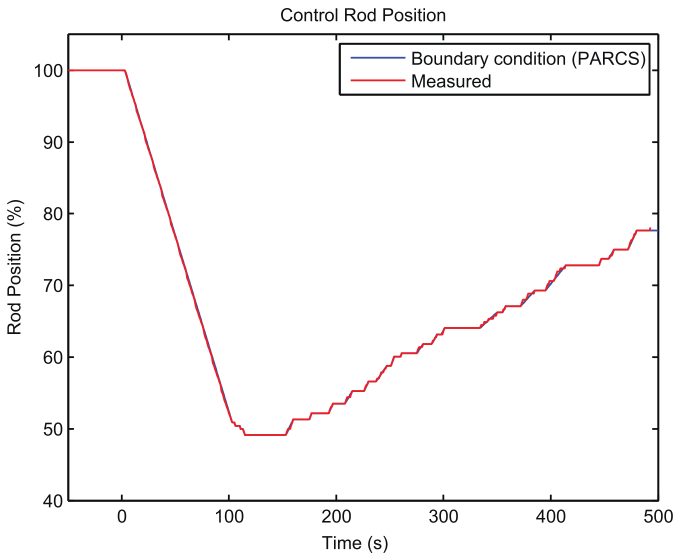


Fig. 6. The position of control rods during the transient.

calculation, the boron concentration is varied to simulate the manual injection performed by the reactor operator. For our coupled simulation purpose, the boron concentration is obtained from RELAP5 which calculates the boron transport and the resulting concentration is subsequently read by PARCS. The change in the boron concentration is triggered by opening the BAT valves to inject high concentration boron. The data of exact measured concentration as a function of time is, however, not available from Ringhals AB. Hence the starting time of injection and the area of the valve opening become the input parameters.

As shown in Fig. 5 the RELAP5 model of the secondary side does not represent a complete loop. Some components beyond the turbine control valves and turbine bypass dump valves as well as the components before the feedwater control valve are excluded. Consequently the feedwater source, the turbine valve and bypasses are treated as boundary conditions.

The component #851 is intended to model the feedwater pump where the pressure and temperature of feedwater are calculated as a function of flow rate in the form of a lookup table. During the whole transient, the temperature of the feedwater varies very significantly, thus obtaining correct values of feedwater temperature during the simulation is very crucial. The feedwater flow rate is determined by the flow area of the feedwater control valves. Moreover, the stem demands of the feedwater control valves are determined by the steam generator level program and flow compensation, which depend on the instantaneous reactor power level.

The steam line connected to the turbine is modeled as a constant pressure boundary condition which equals the upstream pressure of the high pressure turbine. The turbine control valve is an input variable where the stem demand as a function of time is determined by the code user.

On the other hand the steam line connected to the steam dump valve is a time-dependent variable and the pressure boundary as a function of time is determined by the code user. The stam demand of the SDV is calculated from the control system, depending on the average temperature of the primary loop and the reference temperature which varies according to the instantaneous reactor power level.

To perform the simulation of the load rejection test, the following steps were performed:

1. The coupled PARCS/RELAP5 calculation was launched in a PVM environment for ∼300 s to ensure a stable behavior had been reached. The codes were run with “transient” option instead of “steady-state” in order to avoid a premature termination of the calculation.

   The following strategy was applied for achieving steady-state. The calculated heat balance resulted in a particular primary loop flowrate. This mass flowrate was set as the prescribed quantity for the circulation pumps to be reached by variation of the pump speed. The level in the pressurizer was achieved by activating the level control system, which was stabilizing the primary inventory from the chemical and volume control (CVCS) system. On the secondary side, the necessary steam generator level was reached by controling the feedwater flow.
2. The transient coupled calculation was performed subsequently from restarting the last calculated steady-state conditions. This step was run for another 500 s.

   In this step the position of the control rods was described in the PARCS input file, meanwhile the behaviors of the turbine control valve, the pressure of the condenser, the opening of the BAT valves were defined in the RELAP5 input file.

5.2. Steady-state results

By the end of the steady-state calculations, all parameters have reached stable values. Some of the main parameters are presented in Table 2.

Due to the pump speed control system, the reactor coolant flowrate stabilized very rapidly. Only the relative value of this parameter was available in the measured database. Therefore, the corresponding numerical value (expressed in kg/s, instead of %) was determined from the heat balance.

The amount of feedwater and steam is regulated by the level control system of the steam generators. The stabilized values converged to approximately 1646 kg/s which accounts for a discrepancy of around 20 kg/s from the measured values (or less than 1.5% deviation). The average narrow level in the steam generator converged to around 69% which was the control setpoint value. The average pressure in the steam generator also stabilized and gave a very small deviation to the measured data. This shows that the control system implemented in RELAP5 is accurate enough to approach a steady condition.

Following the recommendations in the code User's Guide and Input Requirements (U.S. NRC, 2006), the primary system pressure during the steady state calculation is maintained by a boundary volume connected to the pressurizer to speed up the initialization process as the pressurizer pressure controller has a relatively large time constant. This artificial boundary volume is eliminated at the initiation of the transient calculation and beyond that point the pressure controller will act on the system pressure. The level in the pressurizer is regulated by the level control system. The stabilized value converged with a very good agreement with the measured data. This shows the accuracy of the control system model implemented in RELAP5.

Both the cold leg and hot leg (HL) temperatures stabilized with small deviations from the measured values. Thus, it can be concluded that the overall calculated parameters became stable and reflected the measured plant data with a good degree of accuracy at the end of the initialization process.

5.3. Transient results

The nuclear power generated in the reactor is presented in Fig. 7. Just before the transient initiation, the electric generators were disconnected from the grid, causing the frequency of the plant electric grid to increase, the reactor coolant pump (RCP) to speed up and consequently increasing the power due to moderation. The RCP in the RELAP5, however, is not modeled to be directly connected to

Table 2
Comparison of calculated and measured steady-state parameters.

Parameter	Unit	RELAP5 value	Measured value	Deviation (%)
Average normalized power	%	100.00	99.80	0.020
Total reactor coolant flowrate	kg/s	14,457.84	14,458.00[a]	0.001
Total feedwater flowrate	kg/s	1646.19	1661.72	0.935
Total steam flowrate	kg/s	1645.42	1668.24	1.368
Average narrow range (NR) level in SG	%	68.96	69.31	0.505
Average pressure in SG	bar	61.61	61.57	0.065
Level in pressurizer	%	46.65	46.72	0.149
Pressure in pressurizer	bar	155.11	155.12	0.006
Average temperature in CL	K	556.62	556.49	0.023
Average temperature in HL	K	595.23	595.49	0.044

[a] From heat balance calculation.

the turbine (as the turbine is not explicitly modeled). The increase in the RCP rotation and the increase in the reactor coolant flow is thus not present. Consequently the first peak of the power during the first few seconds after initiation of the transient is not observed in the RELAP5 calculation. A subsequent sudden drop of the power is caused by the increase in the reactor coolant temperature and this behavior is captured quite accurately. However, the second peak of the power, which was caused by the decrease of the reactor fuel temperature, can not be simulated. At this moment the negative reactivity of the control rods overcomes the positive reactivity of the fuel, hence the power keeps going down. Compared to the measured data up to 100 s after the initiation of the transient, the decrease of power has the same slope, indicating the control rods are accurately modeled in PARCS. The time-shifted behavior is mainly caused by the absence of the second power peak. Beyond 120 s after the initiation of the transient, the behavior of the power is determined by a complex interaction between the fuel and coolant temperatures, the insertion of the control rods and the injection of the boric acid into the coolant. In general PARCS manages to reproduce the power during the transient rather well.

The axial power profile before the transient is very similar to the one shown in Fig. 1. During the transient the power profile changes as the power level changes and the control rods were inserted into the core. The location of the power peak shifted as well as its magnitude. The evolution of the axial power profile can be seen in Fig. 8.

The radial distribution of the average fuel temperature is shown in Fig. 9. Four fuel assemblies at the central region of the core experience the highest temperature at the beginning of the transient, while other assemblies in the inner region have a temperature range of 1150–1375 K. The temperatures of the peripheral assemblies are much lower of around 950 K. At 500 s after the initiation of the transient, the core has cooled down with a maximum assembly temperature less then 1300 K.

Fig. 10 presents the time evolution of the axial fuel temperature profiles. The behavior of this time evolution is similar to the axial power profile (Fig. 8). The shifting of the location of the maximum temperature is quite obvious. Please note the stratification seen in this figure is caused by the use of 8 discretized levels in RELAP5, instead of 26 levels as in PARCS.

The radial distribution of the average moderator temperatures at the beginning and end of the transient are shown in Fig 11, while the time evolution of the axial moderator temperature profiles is presented in Fig. 12.

As the frequency of the plant electric grid increases, the TCVs are closing and the majority of the generated steam was dumped to the condensers through bypass lines by opening the SDVs. The temperature and pressure (Fig. 13) on the secondary side then increase when the TCVs are closing. This leads to an increase of the primary coolant temperature, a pressurizer surge (Fig. 14), and consequently to a decrease of the reactor power. The closing of the turbine valves creates a temporary isolation of the steam generators until the SDVs are opening around 2 s later. This isolation induces a sharp drop in the mass flow rate in the steam-lines at the transient initiation. This phenomena is well captured by RELAP5 (Fig. 15). The feedwater control system is thus trying to balance the sudden drop of the steam flow rate by reducing the flow rate of the feedwater. RELAP5 is also able to simulate this behavior, although the flow rate is not as low as in the measured data (Fig. 16).

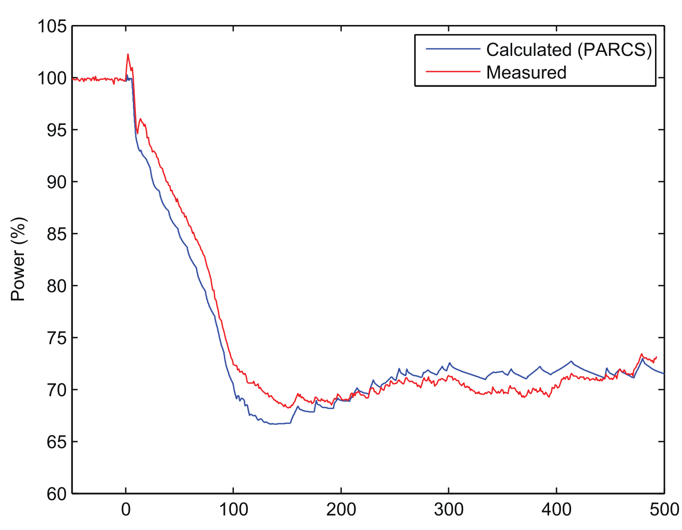


Fig. 7. Nuclear power as function of time during the transient.

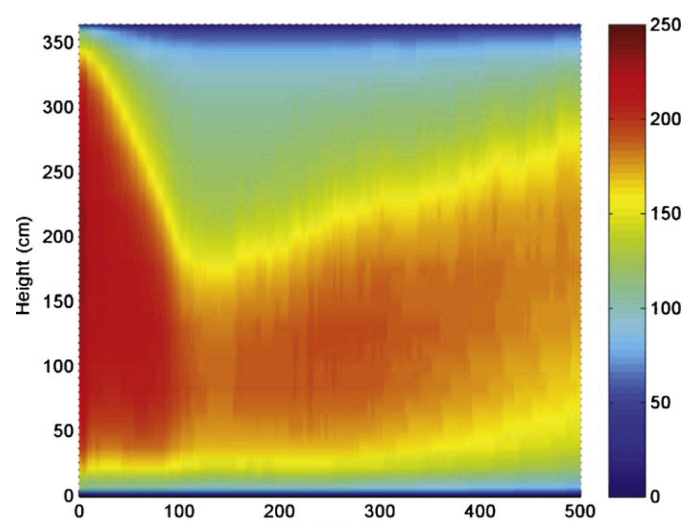
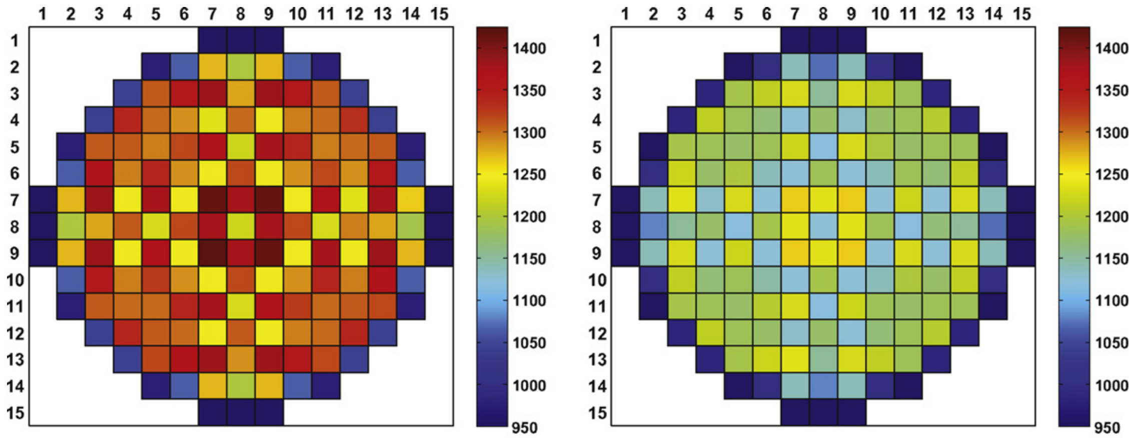


Fig. 8. Time evolution of the axial power profile (in W/cm).



(a) Time = 0 s (b) Time = 500 s

Fig. 9. (a and b) Radial distribution of the fuel temperature in K, averaged along the height of the assemblies.

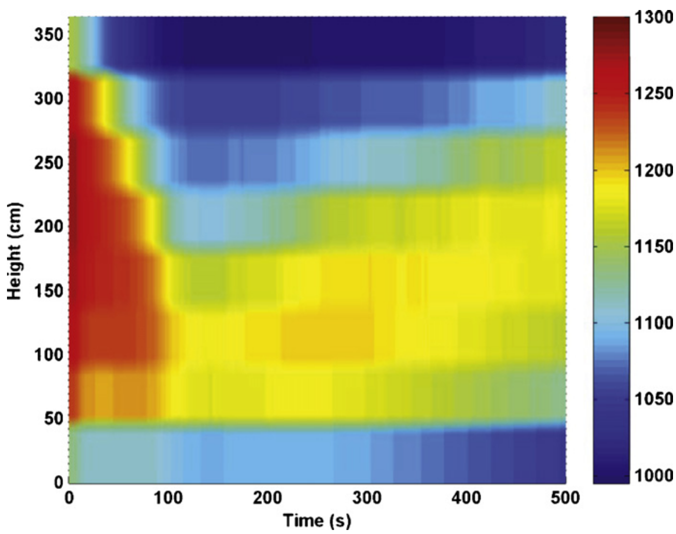


Fig. 10. Time evolution of the axial fuel temperature profile in K.

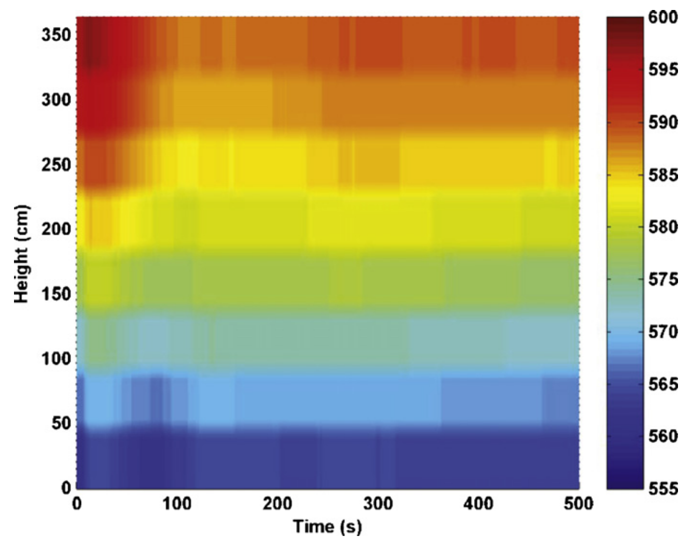
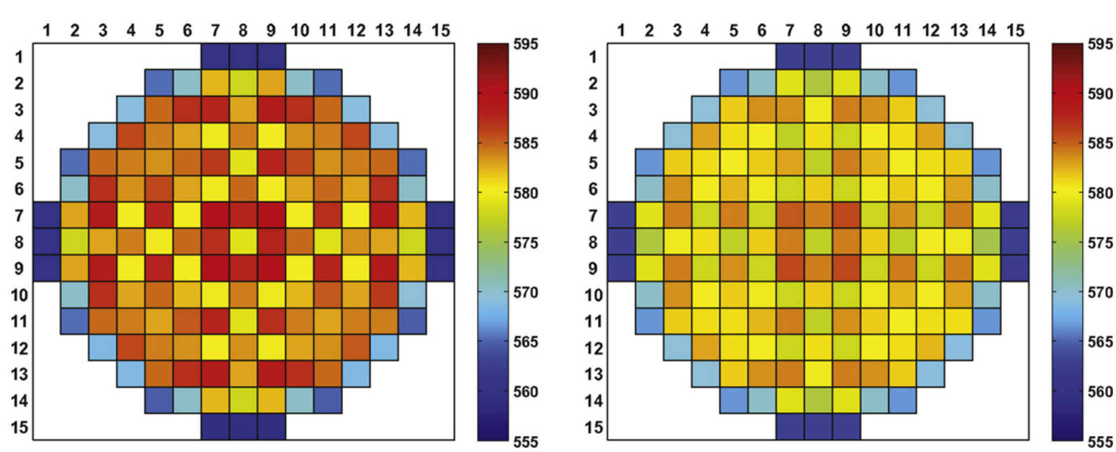


Fig. 12. Time evolution of the axial moderator temperature profile in K.

The consequence of the dump control on the steam-line pressure is even more enhanced in Fig. 13. Temporary isolation of the steam generators resulted in a large jump in the pressure. During the period of the full dumping, the pressure is declining. When the TCVs are opening, the SDVs partially opens and thus repressurization occurs. The input signal for controlling the SDVs opening or closing is a function of the primary side loop average temperature. RELAP5 could simulate the repressurization of the secondary side quite good.



(a) Time = 0 s (b) Time = 500 s

Fig. 11. (a and b) Radial distribution of the average moderator temperature in K, averaged along the height of the assemblies.

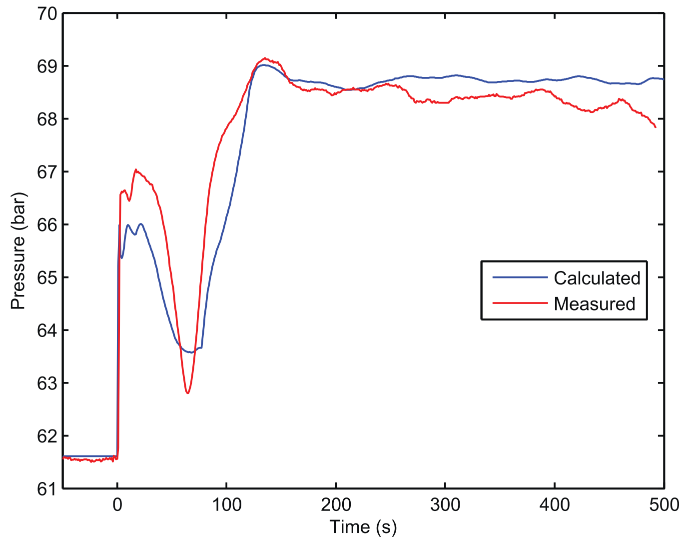


Fig. 13. Average steam line pressure during the transient.

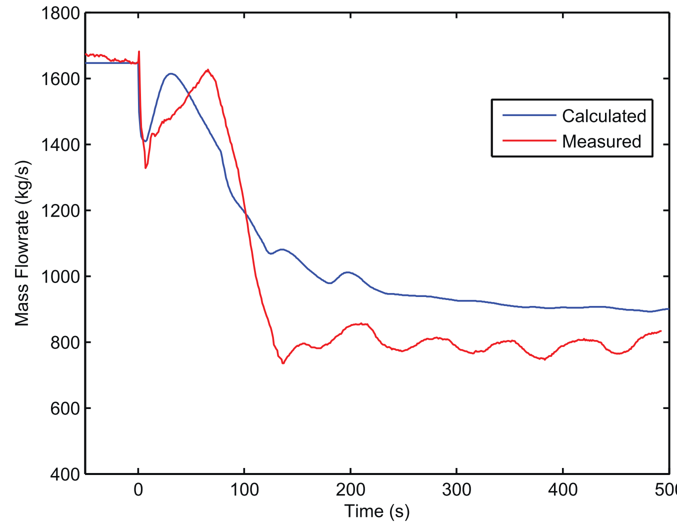


Fig. 16. Total feedwater flowrate during the transient.

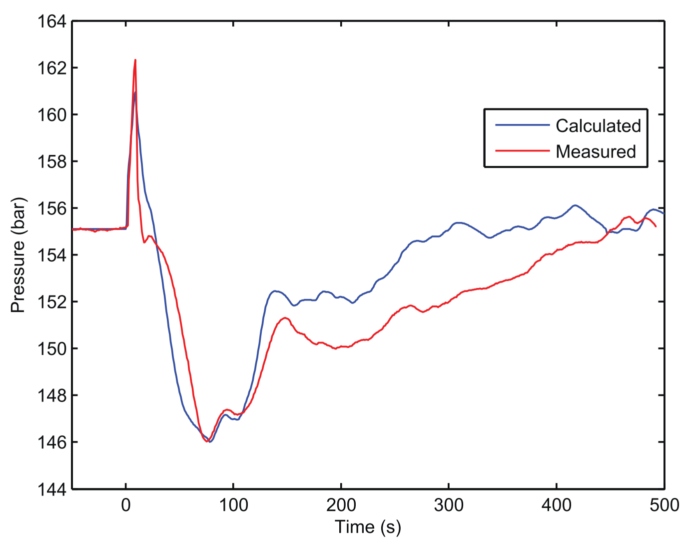


Fig. 14. Pressurizer pressure during the transient.

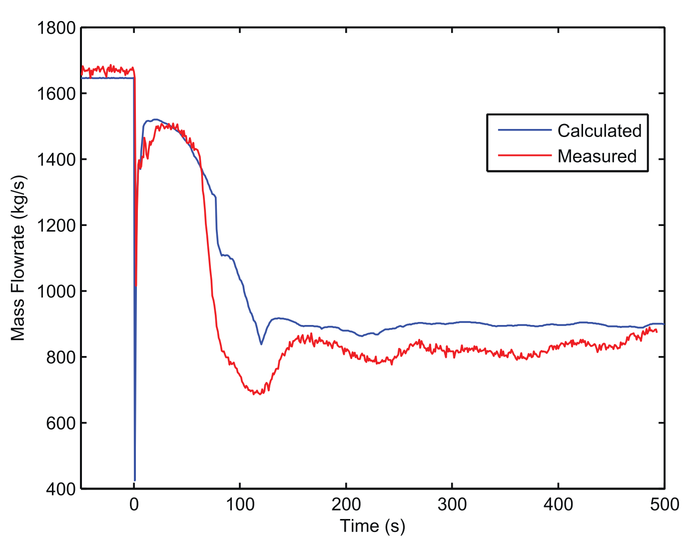


Fig. 15. Total steam flowrate during the transient.

In the primary side, the continuous decrease of energy during the dump process is well represented in the pressurizer pressure (Fig. 14). The transient initiation resulted in a large jump followed by an uninterrupted decrease of the pressure. The peak of the pressure reached 162.4 bar and triggered the opening of the pressure-operated relief valve (PORV) which has the setting point of 162 bar. However, this event did not occur in the simulation as the maximum pressurizer pressure was roughly 161 bar. Subsequently repressurization of the pressurizer started at approximately 90 s. The timing was well captured by the code. After 150 s RELAP5 underpredicts the pressure with a maximum difference of around 2 bar. The calculated line, however, remained basically parallel with the measured one.

In the hot leg temperature, a minor initial peak was observed in the hot-leg temperature, followed by a cooling and then stabilization to a roughly constant value. Again, RELAP5 was able to reproduce the temperature increase and decrease, but settled to higher temperature with a maximum difference of around 2 K (Fig. 17).

The general trends of the cold-leg temperatures are qualitatively similar to the steam generator pressure curve demonstrated before. As depicted in Fig. 18, the cold-leg temperature dropped

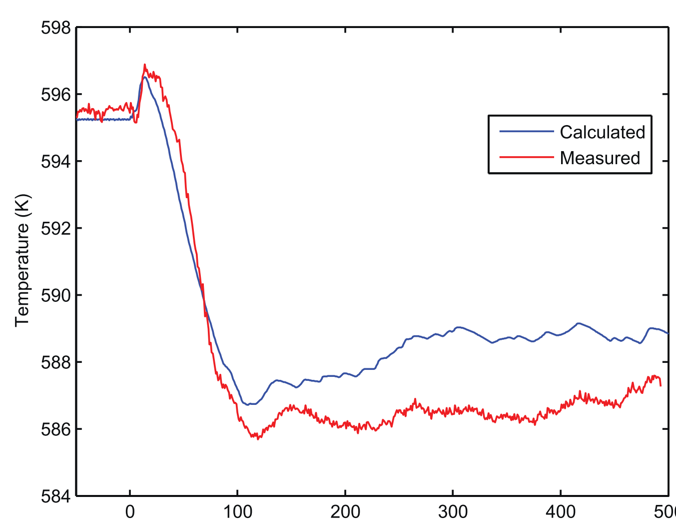


Fig. 17. Average hot-leg temperature during the transient.

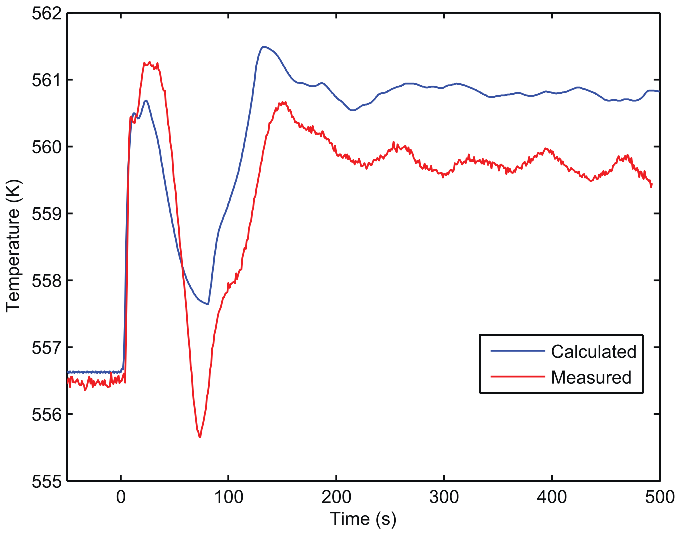


Fig. 18. Average cold-leg temperature during the transient.

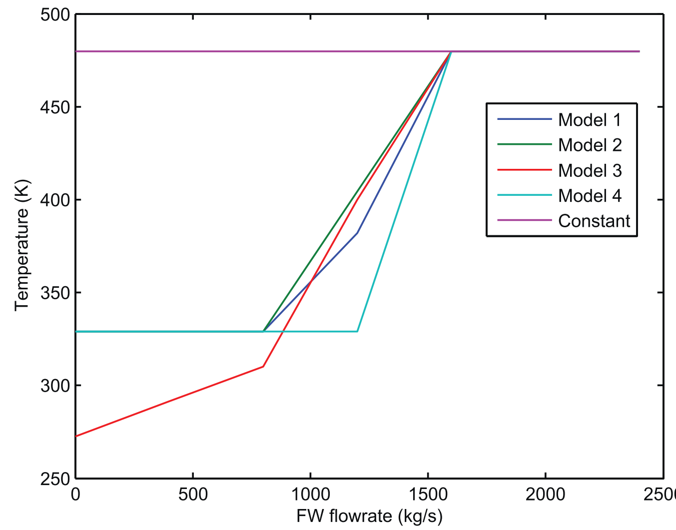


Fig. 19. Feedwater temperature model as a function of FW flowrate for parametric study.

below 556 K at the end of full dump valve opening. This did not happen in the code calculation, therefore a remaining discrepancy of approximately 1–1.5 K could be observed in the rest of the transient.

6. Parametric study

As indicated in Section 5.1, the current RELAP5 model does not include some components between the turbine control valves and the feedwater control valve in the secondary loop. The feedwater source, the turbine valve and bypasses are thus treated as boundary conditions. The data required for setting the boundary/initial conditions are however not immediately available from the plant. This lack of data may become one source of the uncertainties to the results (see e.g. the discussion by Petruzzi and D'Auria, 2008 and D'Auria et al., 2006). The following sections discuss the results of parametric studies on the boundary conditions which give large uncertainties.

6.1. Temperature of feedwater

During the load rejection transient, the temperature of the feedwater decreases at the beginning and then settles down to a constant value. The plant data of the feedwater temperature is available, but it is not used as an explicit boundary condition in the RELAP5 model. The feedwater temperature is modeled indirectly as a function of the feedwater flow rate. By doing so the temperature of the feedwater can not be determined beforehand as the feedwater flow rate is regulated by the control system which depends on the power generation in the reactor and the production of steam in the steam generator. Fig. 19 represents various models of feedwater temperature as a function of feedwater flowrate. The constant temperature model is the model used in the original Studsvik EcoSafe input (Eriksson, 1994). The other four models are determined by observing the trend of the feedwater temperature and the feedwater flow rate in the actual plant data during the transient.

The feedwater temperature along the transient for different models is shown in Fig. 20. The transition of the actual temperature from the plant data (red line) runs smoothly from 480 K to 329 K. However, all models but the constant temperature model show a dip in temperature which ranges from 440 K to 460 K during the beginning of the transient. This behavior is understandable because the feedwater flowrate is reduced to balance the sudden drop of the steam production. Model 4 reaches the actual steady value at earlier time while Model 2 gives the largest discrepancy of around 25 K.

Fig. 21 shows the influence of various feedwater model to the feedwater flowrate along the transient. It can be seen clearly that the initial dip of the flow is not sensitive to various fedwater temperature models. However in later time, all models produce discrepancies. The constant model present the largest discrepancy, indicating that the original Studsvik EcoSafe model is not applicable to the load rejection transient. The other four models show similar behaviors, i.e. decrease and settle to a quasi-steady value. Model 4 presents the best result, having a discrepancy of around 50 kg/s. These results imply that a better agreement with the measured plant data may be achieved provided that an accurate feedwater temperature is given.

6.2. Opening area of turbine control valve

At the beginning of the transient, the TCVs are closed to prevent an overspeed of the turbine rotation and later as the turbine frequency regulator has adjusted the number of the rotation, the

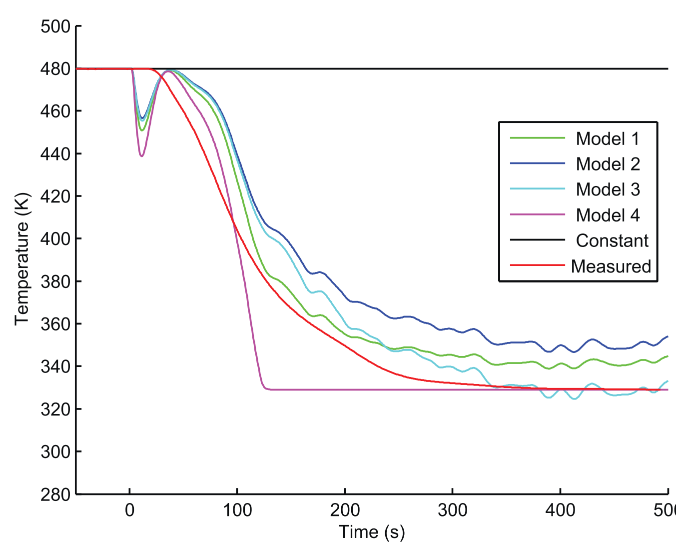


Fig. 20. Feedwater temperature during the course of the transient. (For interpretation of color in the artwork, the reader is referred to the web version of the article.)

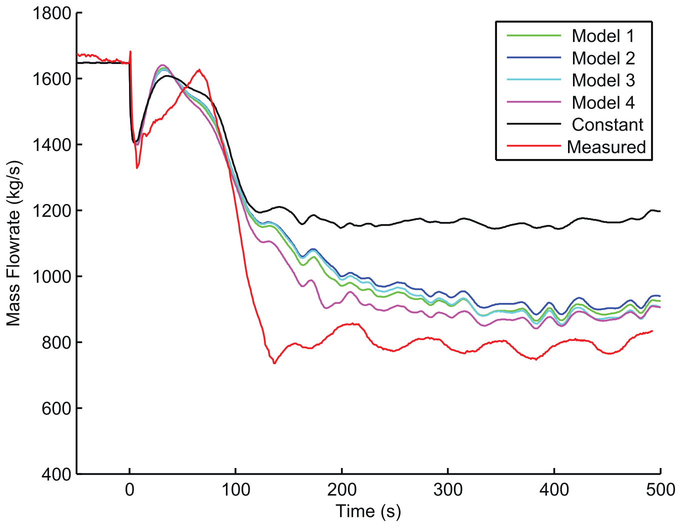


Fig. 21. Feedwater flowrate during the course of the transient.

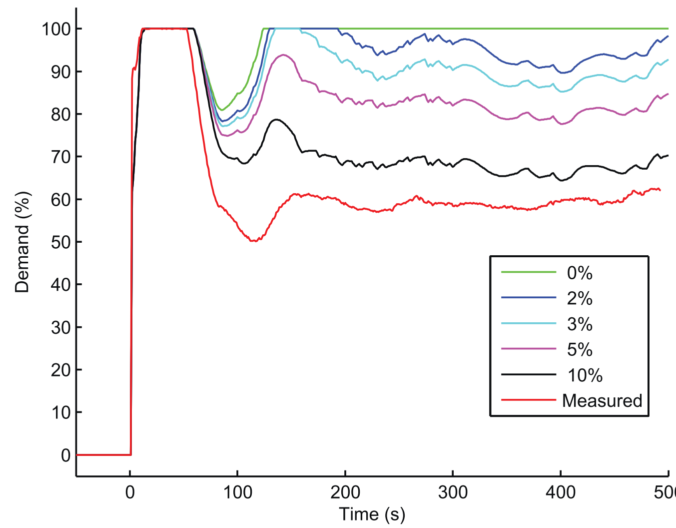


Fig. 23. Steam dump demand at various opening size of turbine control valve. (For interpretation of color in the artwork, the reader is referred to the web version of the article.)

TCVs are partially opened. The opening of the valve is governed by a complex turbine control logic system and thus it is just simply modeled as a boundary condition in the RELAP5 model. There is no clear information on the opening size of the TCV, thus this lack of data may contribute to the uncertainties of the results as well.

The influence of various opening areas to the average steam line pressure can be seen in Fig. 22. The larger the area, the lower steady value the pressure settles to. Opening the valve between 3% and 5% seems to give a good agreement with the measured pressure. Such a condition, however, does not reflect on other parameters. Fig. 23 shows the demand of the steam dump valve, which corresponds to the bypassed steam flow to the condenser. If the TCV remains closed during the transient, all steam will flow through the dump valve, and there will be 100% steam demand, as indicated by the straight green line. When the TCV is partially opened, the steam demand of the dump valve will be less than 100%. At 3% and 5% of TCV's opening size, the steam demands do not come close to the measured data. The least discrepancy is obtained when the TCV opening is 10%.

6.3. Time to maximum steam dump pressure

When the TCVs are temporarily closed, high pressure steam is bypassed to the condenser through the SDV. This will temporarily increase the pressure of the condenser before it settles down to a lower value when the opening of the TCVs is regulated. As the SDV is connected to the condenser, the downstream pressure will also vary during the transient. This information is not known directly, hence several models of pressure transient are implemented as boundary condition.

Fig. 24 presents the influence of different models to the average steam line. The models are represented here as the time to achieve the maximum SDV pressure. As can be seen in Fig. 24, the dynamics of the steam pressure is sensitive to the transient of the SDV pressure. An earlier time to reach the maximum will overshoot the actual steam line pressure. A later time to reach the maximum will dampen the oscillation, but the response is delayed during a pressure buildup.

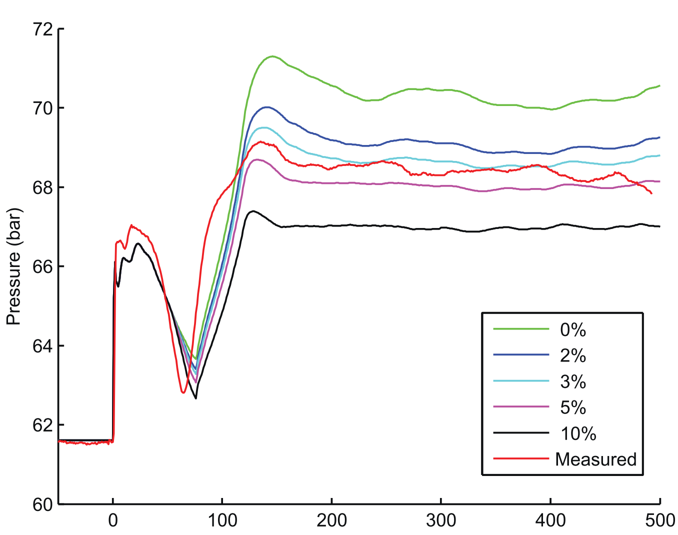


Fig. 22. Average steam line pressure at various opening size of turbine control valve.

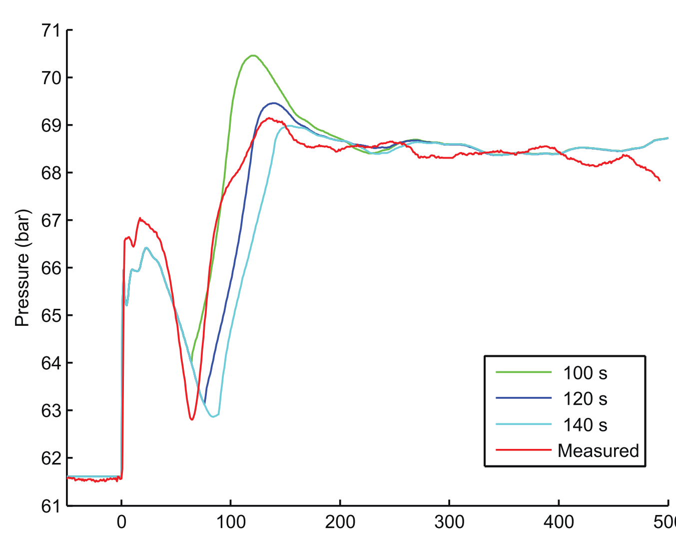


Fig. 24. Average steam line pressure at various time to achieve maximum dump pressure.

7. Conclusions

A validation study has been presented in this article in order to test the applicability of a coupled neutron kinetic/thermal-hydraulic model for the Ringhals-3 NPP. Verification of the applied components and techniques against a realistic measured database has been performed.

It has been shown that the coupled codes in general are capable of reproducing all the general trends in the load rejection transient. Matching of the recorded plant parameters and the code calculation is good in most cases. Some discrepancies, however, are observed, indicating a need of further analysis and revision.

The transient may essentially be split into two parts. The first part is basically the dumping with fully open dump valves. This period is relatively well reproduced by the RELAP5 code. The second period starts with a sharp re-pressurization of the secondary side, when the dump control system actuates the dump valves to close. Larger discrepancies are observed in this period, indicating that further efforts are necessary. The lack of real plant data in some input parameters may contribute to these discrepancies. These parameters include, for instance, the timing and the magnitude of opening the high pressure turbine control valve. In a real plant operation, the opening of this valve is actuated and controled by a sophisticated turbine control system, which may not be easy to be modeled in RELAP5.

Another source of discrepancies is the lack of realtime data of boron concentration in the coolant. In the coupled model, the change in boron concentration is calculated through the boron transport model in RELAP5, and not directly embedded in the PARCS model. Hence an accurate model of the onset, the duration and the magnitude of the BAT valve opening is necessary.

It should be mentioned as well that inaccuracy of the feedwater source model may contribute to the discrepancies. In a real situation, the feedwater temperature decreases during the load rejection transient because the steam flow to the feedwater heaters is terminated. However, modeling an accurate feedwater temperature along the transient is quite challenging. The feedwater temperature is not modeled as an explicit boundary condition to the system, but rather as a function of the feedwater flow rate, which is determined by the control system through the opening of the feedwater valve. Just to demonstrate the complexity of this problem, the demand for opening of the feedwater valve is an indirect dependence on the reactor power level, which obviously also depends on the accuracy of the boron transport.

Improvements to the coupled PARCS/RELAP5 model can be performed for future works. Closing the secondary loop by including turbines, condenser, feedwater heaters and other components could be done to get better results and better insight about the ongoing process. This activity, however, is quite challenging as the number of the components involved will increase to a large extent with their corresponding boundary and initial conditions. From RELAP5 point of view, modeling of turbine should be performed cautiously as no user experience exists applying the turbine component beyond that in code developmental assessment so far (U.S. NRC, 2006).

The parametric study on the boundary conditions shows that some parameters are sensitive to the given boundary conditions. Further evaluation by incorporating uncertainty and sensitivity methods will be useful to determine the uncertainty band.

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