VALIDATION OF RELAP5/MOD3.3 AGAINST A LOAD STEP TRANSIENT AT RINGHALS 4 NPP

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ABSTRACT

The Unit 4 of the Ringhals Nuclear Power Plant has recently undergone a large component replacement project with installation of new steam generators and a pressurizer, targeting a power uprate. A series of startup and maneuverability tests has been performed, mainly focusing on evaluation of the system responses for various perturbations. The subject of the present numerical analysis is a test with a ± 10 % steps applied in the load. The data collected during test provided a good opportunity for validation of the full plant model, which was prepared recently for the RELAP5/Mod3.3 Patch04 computer code, with incorporation of the new component models.

The paper introduces the test procedure and shows an overview on the key parameters that are utilized as initial and boundary conditions. Strategies applied for achievement of steady-state conditions are addressed in the document. Furthermore, the paper summarizes the results of the validation study using the transient data to simulate the startup test. It has been proven that the stand-alone RELAP5 thermal-hydraulic model is capable of reproduction of the key features and events of the test. Sufficiently good agreement has been achieved between the measured and simulated thermal hydraulic parameters, already in its current stage of model development. On the basis of successful verification at the original power, it is expectable that the new Ringhals 4 model will be able to predict the fluid conditions in other types of transients, even at uprated conditions.

KEYWORDS
RELAP5 validation, Ringhals 4, load step transient, full plant model

1. INTRODUCTION

1.1. Background

Ringhals 4 (R4), a 3-loop Westinghouse-type NPP was commissioned in 1983. It went through a number of extensive modifications and modernizations in 2011. Within the project of “FREJ/TURBO”, the steam generators (SGs), the pressurizer (PRZ), and large parts of the turbine plant have been replaced. Refurbishment of these components affected also the relevant control and protection systems. The new nominal thermal power of the unit is 3300 MW in the current test-operational state. However, the permitted maximal power is still limited to 2783 MW (84.3 %), until the unit obtains the license for the final upgrade.

The Swedish Radiation Safety Authority (SSM) has received the license application for the power uprate. Chalmers University has been collaborating with SSM for many years, acting as a Technical Support Organization in the decision making process of the power uprates of the Swedish PWRs. Chalmers is also contributing to the domestic nuclear safety projects with evaluation of the consequences of the increased power by deterministic numerical system code calculations.

1.2. Purposes of the Test

To ensure the continued safe operation of the plant, several tests have been conducted after the start-up. One of the first tests was a ± 10 % load step test (QUATTRO/R4 – QP-101), which is the subject of the present study. The main purpose of the current start-up test was to verify that the unit, including the recently installed components, together with their automation and control systems, are able to handle a perturbation of the power in a satisfactory manner. Specifically, two key issues were highlighted in the investigations:

- Checking the ability of the rod control system to adapt the reactor power according to the turbine power needs, without activation of the steam dump.
- Collecting data on the dynamics of SG and PRZ level changes due to the temperature variations in the reactor coolant system (RCS).

The magnitude of the power step was carefully set to ± 10 % because this change is large enough to generate a challenge for the control systems, but still it remains under the level, which would trigger any setpoint of protection system, such as a reactor scram or opening a safety valve. A detailed description of the entire test procedure is given in [1].

1.3. Objectives of the Model Verification

A new stand-alone RELAP5 computer code model has been built in order to simulate the entire unit 4 at Ringhals by utilizing the experience gained with unit 3 modeling. Comprehensive descriptions of the full-plant models of R3 and R4 are reported in [2] and [3], respectively. The new code input includes the replaced component models of the EPR-type SGs and the PRZ models, which were prepared on the basis of technical details provided by the designer. The limitations and setpoints specified by the utility [4] reflect the nominal parameters for 100 %, i.e. 3300 MW thermal power, and the model is built accordingly. However, the operation at a lower power needed some modifications and adaptations in the input.

The present validation study is a part of the process in which the ultimate goal is to deliver a thoroughly verified, multi-purpose thermal hydraulic model of R4 that is applicable for basically any type of transient analysis. Within the progress of the model development, the load step start-up test of QUATTRO/R4 – QP-101 gave the first opportunity for verification of the model performance against a real plant transient.

2. THE TEST PROCEDURE

2.1. Part A: Power Decrease

Preparation for the test started with a gradual decrease of the thermal power from 84.3 % to approximately 80 % on 2011-12-14 at 17:00. Part A of the real test began at exactly 17:46, when the power of turbine system T41 was decreased from 472 MW to 351 MW. This is equivalent to a 12.2 % drop in the expected total nominal turbine power. Decrease of the reactor power was achieved by automatic insertion of the control rods with maximal speed. The effect is visible on the signal of the neutron flux indicator, which is reduced from 81 % to 72 % (Figure 1). Reading from the plant instrumentation, the thermal power was decreased by approximately 305 MW (i.e. 10.9 %).

2.2. Part 2: Power Increase

The second part of the test started by restoring the initial power on the generator. This meant that the turbine power T41 increased from 352 MW to 472 MW (equivalent to 12.1 % of the total nominal turbine power). The consequence of pulling out of the control rods was an increase of the neutron flux signal from 72.5 % to 79.5 %. The plant instrumentation indicated a thermal power increase of 311 MW.

2.3. Rod Position vs. Power

As Figure 1 shows, position of the control rods and the neutronic power are strongly correlated. However, determination (and modeling) of the actual thermal heat source for the reactor, just purely based on these parameters is not straightforward. This is due to the fact that the rod drive mechanism is partly controlled by the impulse chamber pressure. The difference between the turbine power and reactor power leads to a large "power mismatch" which activates the rod regulation and the rods are inserted with maximum speed. The system made an overshoot of 2 % before stabilization. The effect was that the control rods were inserted initially a bit too far, therefore they had to be pulled out slightly when the impulse chamber pressure increased, which could also be attributed to the reactor power increase.

During the load increase, value of the "power mismatch" became strongly negative, consequently the rods were pulled out with maximum speed. After this last step, the neutronic power was also influenced by boron injection and the growth of xenon. These factors had an effect of the reactor thermal power, as well.

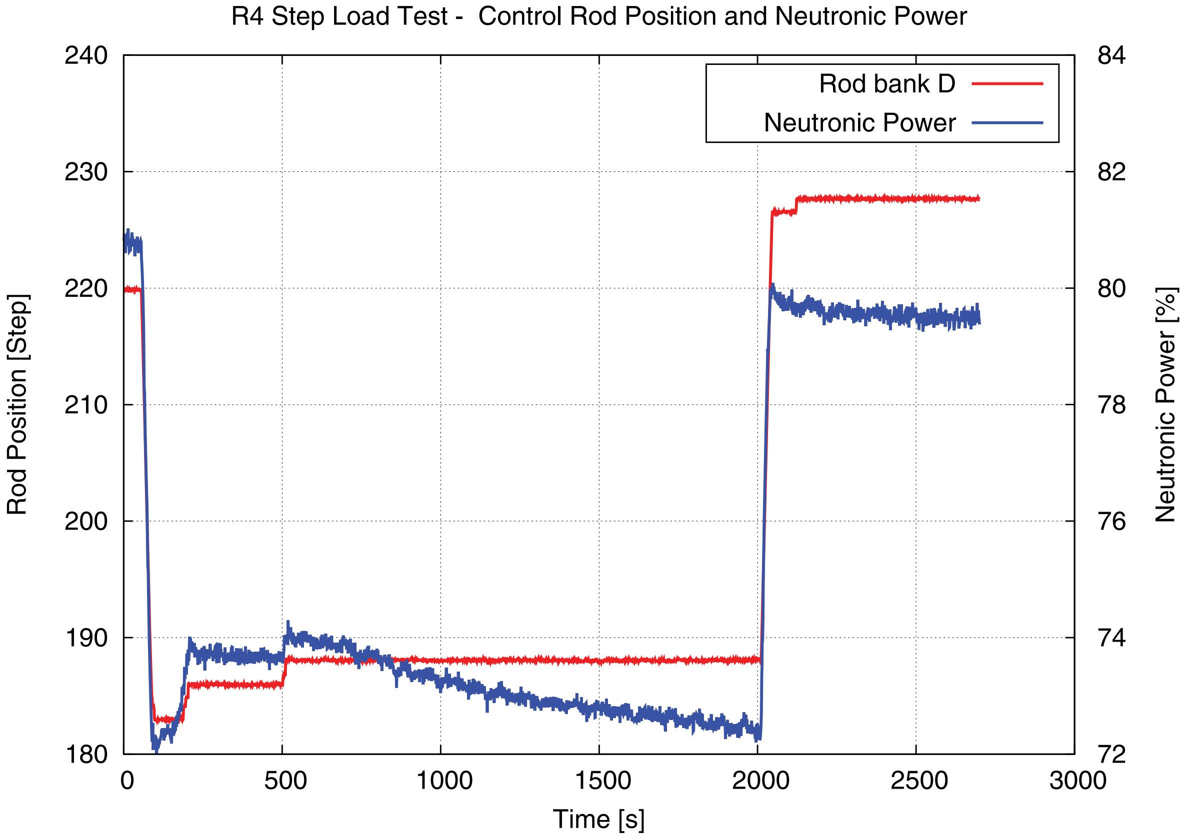

Figure 1 Control rod position and neutronic power

3. THE RELAP5 MODEL OF RINGHALS 4

The current RELAP5 model of the R4 was built on the basis of the Ringhals 3 (R3) input developed at the Department of Nuclear Engineering of Chalmers University of Technology. Since the layouts of the R3 and the R4 units are similar, the main intention was to keep as many elements from the validated R3 model as possible. However, the steam generators and the pressurizer were replaced with the new AREVA design components in 2011. The structures of these parts, especially the SGs are significantly different from those that are installed in R3. Consequently, building a new full-plant model became necessary.

The current R4 model represents the "state-of-the-art" and the practical experiences that have been gained in the modeling process over the last few years. In the core region, each of the 157 fuel assemblies is modeled individually (Figure 2), both for the hydrodynamics and for the heat structures. Axially, the active core was discretized into eight levels.

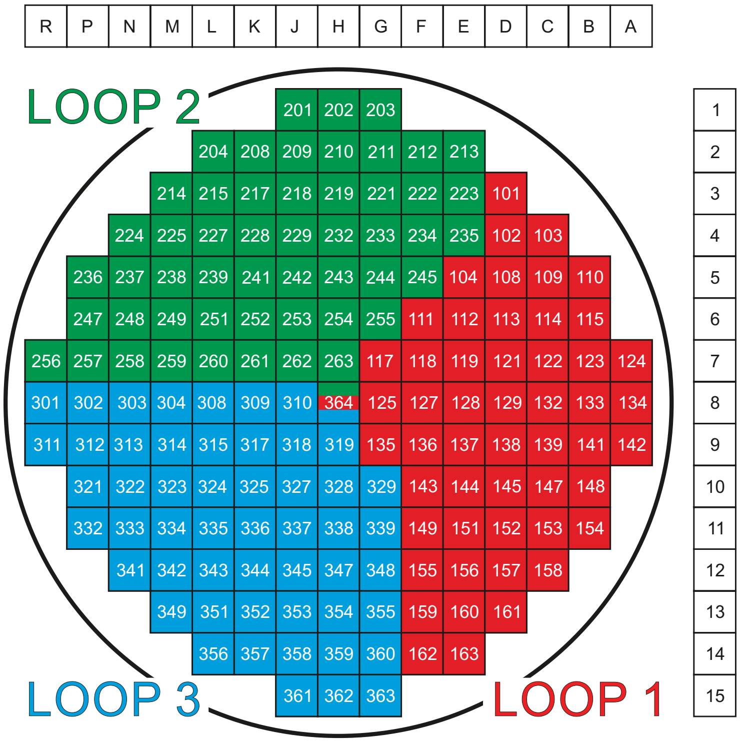

Figure 2 Radial discretization with numbering of the fuel channels

The core inlet and outlet needed special considerations due to a limitation in RELAP5, that a branch component may be connected to a maximum of nine 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 (Figure 3).

It is essential that some specific phenomena, for instance an asymmetric behavior of the loops can be properly captured. For this reason, the downcomer and the core are split into three parallel channels in order to retain the 3-loop structure of the primary side even within the reactor pressure vessel. It is also important to distinguish between the coolant flowing through the fuel assemblies that are heated, and the remaining part of the flow bypassing the core. Thus, the core has been extended with altogether three 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.

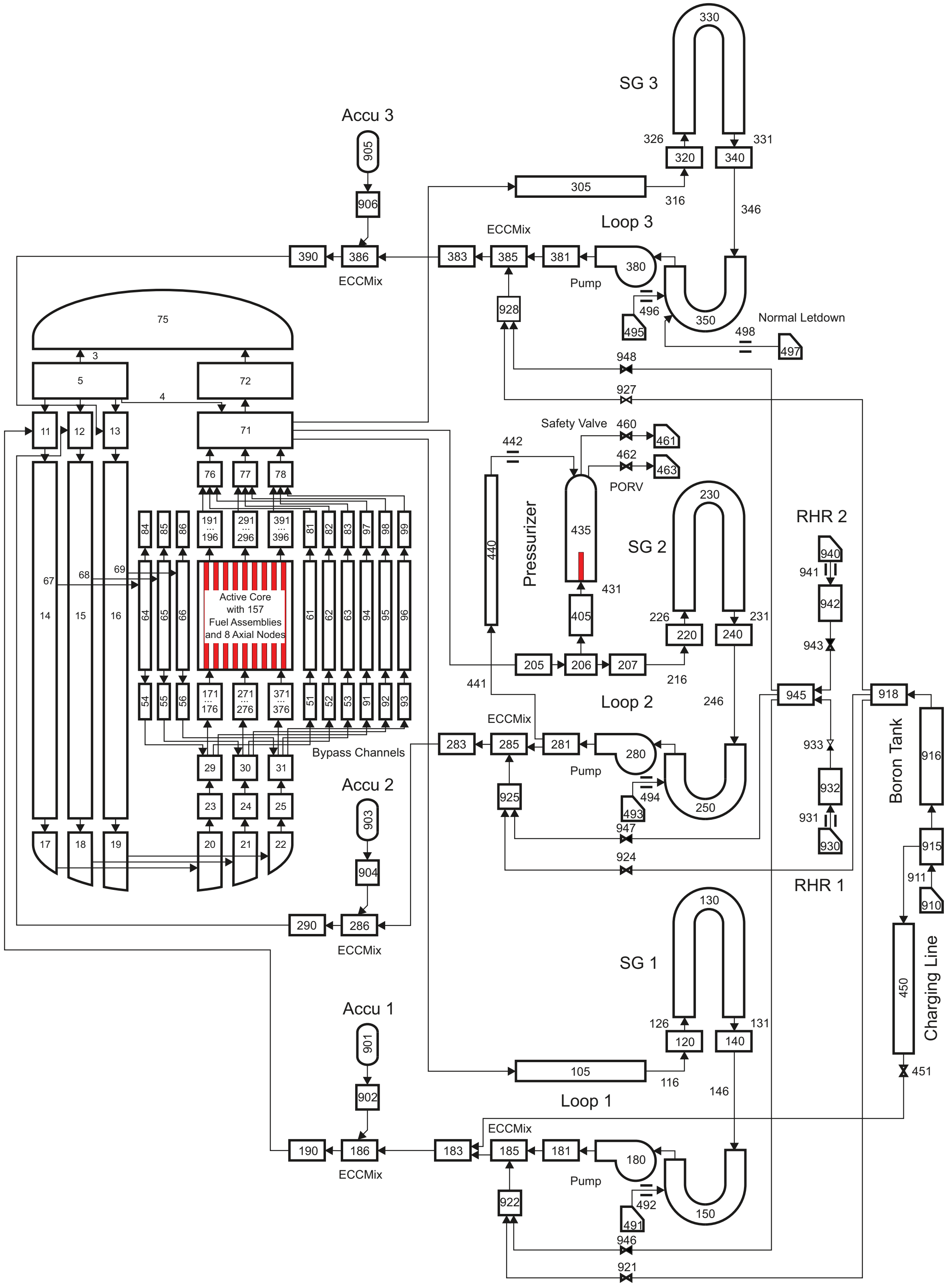

Figure 3 Nodalization of the primary side

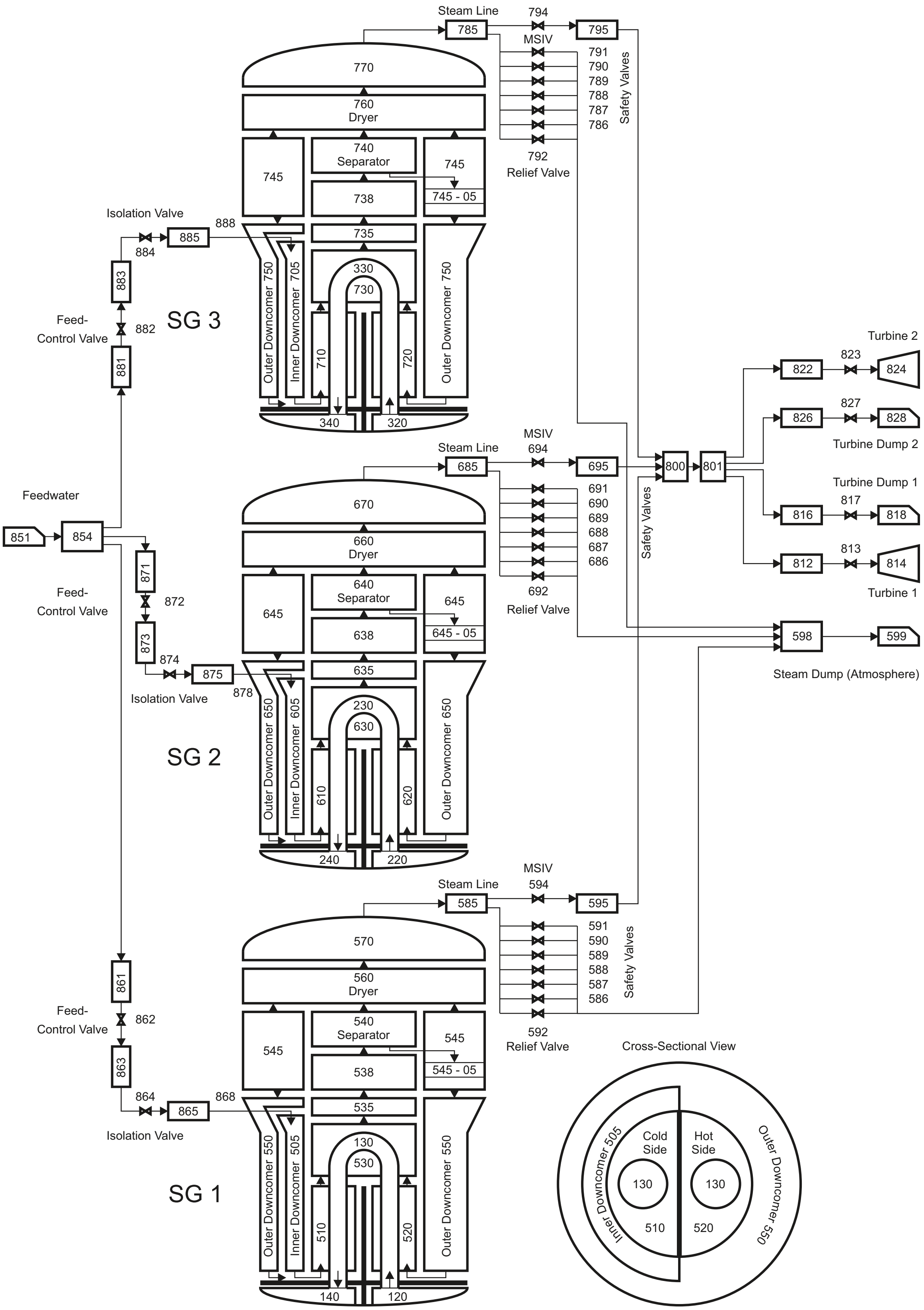

Figure 4 Nodalization of the secondary side

4. ANALYSIS OF THE LOAD STEP TEST

4.1. Achieving Steady-State

4.1.1. Heat balance calculation

It is an elementary requirement that a simulation should reach perfect steady-state before initiating a transient calculation. Looking at the measured plant data, this goal seems to be theoretical because even in the most "stable" operation of a plant, in fact, it is just in a quasi-steady-state condition. Nevertheless, the best possible approach to determination of a consistent dataset is to perform a short hand calculation, using averaged data over the available length of the stable operation prior to the test. With determination of the water properties, this dataset can satisfy the heat balance equations, which will be entered to the RELAP5 input as initial condition.

The heat balance calculation helps also to determine those parameters that are not directly available from the measured database. For instance, the loop flowrate is recorded in percent but the reference (100 %) value is not accurately known. Still, it is necessary to enter the flowrate data in SI units (i.e. kg/s) into the RELAP5 input deck.

With assumption of symmetrical behavior of the loops, and averaging the measured feedwater flows and steam flows, the following simplified heat balance can be set up between the secondary sides and the primary sides of the steam generators:

<u>Secondary side</u>

$$\begin{aligned}
\dot{m}_{FW} &= 460.89 \text{ kg/s} \\
\mathrm{T}_{FW} &= 475.81 \text{ K} \\
\mathrm{h}_{FW} &= 0.8662473 \cdot 10^{6} \text{ J/kg} \\
\dot{m}_{steam} &= 467.26 \text{ kg/s} \\
p_{steam} &= 65.575 \text{ bar} \\
\mathrm{h}_{steam} &= 2.77818 \cdot 10^{6} \text{ J/kg} \\
N_{SG} &= \bar{\dot{m}}(h_{steam} - h_{FW}) = 887.280167 \text{ MW} \\
\sum_{i=1}^{3} N_{SG,i} &= 3 \cdot N_{SG} = 2661.841 \text{ MW}
\end{aligned}$$

<u>Primary side</u>

$$\begin{aligned}
p_{PRZ} &= 154.7 \text{ bar} \\
T_{HL} &= 590.69 \text{ K} \\
h_{HL} &= 1.4381 \cdot 10^{6} \text{ J/kg} \\
T_{CL} &= 557.07 \text{ K} \\
h_{CL} &= 1.2528 \cdot 10^{6} \text{ J/kg} \\
\dot{m}_{loop} &= \frac{N_{SG}}{h_{HL} - h_{CL}} = \frac{887.280167 \text{ MW}}{1.4381 \cdot 10^{6} \text{ J/kg} - 1.2528 \cdot 10^{6} \text{ J/kg}} \\
\dot{m}_{loop} &= 4788.34 \text{ kg/s}
\end{aligned}$$

4.1.2. Strategies to reach steady-state

When the desired initial conditions were determined and entered to the input mostly as setpoint values, special strategies were applied to reach steady-state, which took 4000 s. The calculation was performed by letting the relevant control components to be actuated. The following arrangements were made in the steady-state run with the code:

- The core heating power is kept constant by the user-given heat source data in the fuel bundles. The additional heat generated by the pumps is taken also into account.
- The setpoint value of the primary side pressure can be requested by the user. The PRZ pressure control system regulates the pressure, with variation of the power of the proportional or the on/off heaters or with spraying.
- The desired primary side level is a function of the average loop temperature. The level setpoint is achieved by the PRZ level control system. Filling of the coolant is provided from the chemical and volume control system (CVCS).
- Concerning the primary side loop flowrate, the pump speed is controlled by monitoring the difference between the actual loop flowrate and the prescribed value. The pump rotation is adjusted automatically by the controller. (This is only a feature of the RELAP5 model. The loop flow controller is not physically present at the plant).
- The secondary side pressure is given by a constant boundary condition. The turbine model is a simplified time dependent volume with a pressure boundary at saturated steam.
- The levels in the SGs are maintained by a dedicated level control system. Deviations between the feedwater flows and the steam flows are sensed, and the level is compared to the defined level program, which is always a function of the actual reactor power. The feedwater control valve (FCV) is then actuated by modifying its cross-sectional area with injection of the proper amount of feedwater.
- The feedwater temperature is constant in the steady-state search run. However, the model allows the temperature of the fluid to be a function of time in a transient.

Features of the steady-state run are summarized in Table 1.

Table 1 Features of the steady-state run

Parameter	Unit	Desired	Achieved	Parameter in RELAP5	Setpoint Setting
Thermal power	MW	2661.84	2662.00	cntrlvar-001	cntrlvar-001
PRZ Pressure	Pa	15470285	15470741	p-435120000	cntrlvar-400
PRZ Level	%	39.53	39.69	cntrlvar-430	cntrlvar-430
Loop Flowrate	%	4788.34	4726.62	mflowj-180010000	cntrlvar-180
Hot Leg Temp.	K	590.69	590.68	tempf-120010000	automatic
Cold Leg Temp.	K	557.07	557.03	tempf-140010000	automatic
Average Loop Temp.	C	300.69	300.80	cntrlvar-434	automatic
SG Pressure	Pa	6557548	6571872	p-570010000	p-814, p-824
SL Pressure	Pa	6486870	6489701	p-585050000	p-814, p-824
FW Flow	kg/s	460.89	460.41	mflowj-868000000	automatic
Steam flow	kg/s	467.26	460.34	mflowj-594000000	automatic
SG Level	%	66.20	66.20	cntrlvar-502	cntrlvar-503

4.1.3. Steady-state results

As it is demonstrated in the following plots (Figure 5 … Figure 10), the model was able to bring the entire system into a stable, steady-state condition. All the parameters have converged very close to the desired values. These quantities satisfy the heat balance equations with a good accuracy.

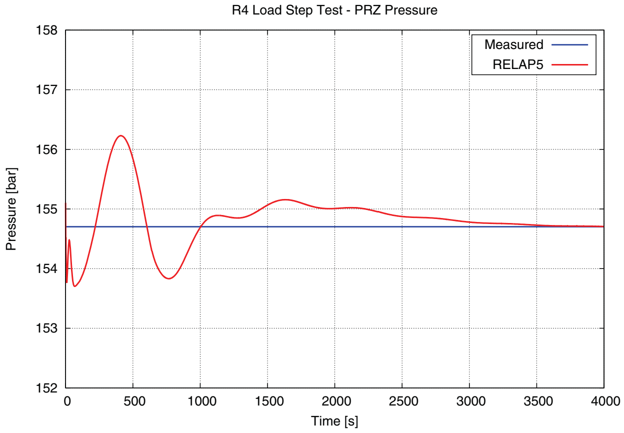

Figure 5 PRZ pressure

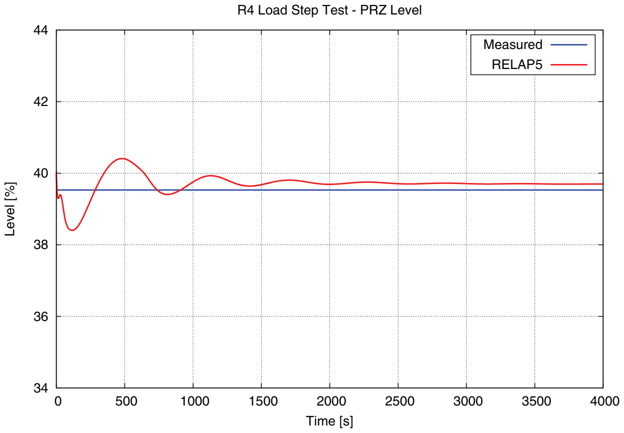

Figure 6 PRZ level

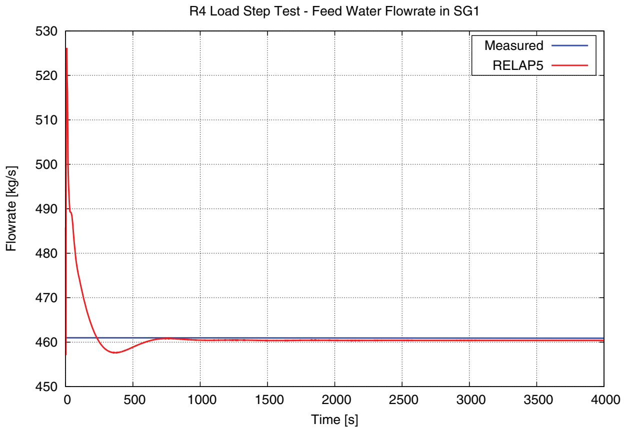

Figure 7 FW flowrate

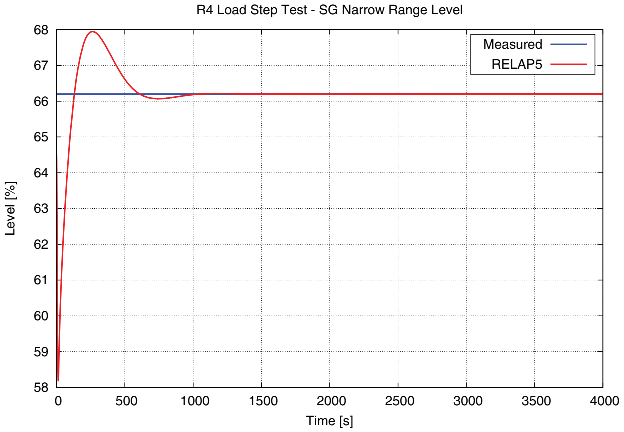

Figure 8 SG level

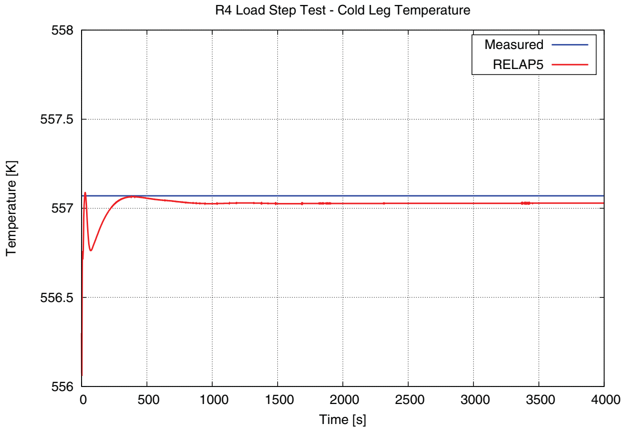

Figure 9 Cold leg temperature

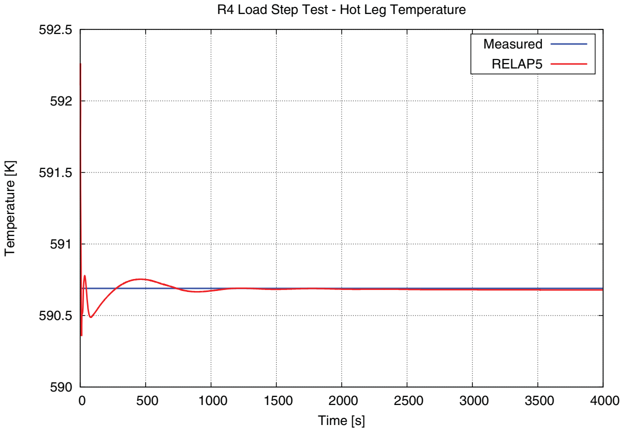

Figure 10 Hot leg temperature

4.2. Analysis of the Transient

It is important to emphasize that the simulation used a stand-alone model of R4 in the current study. It means that only the thermal-hydraulic features of RELAP5 were applied, without any internal neutronic feedback or without coupling to an external neutronic code, such as PARCS. (There are intentions to prepare a neutron kinetic model of unit 4 in the future). Note that with changing coolant temperatures, the density, the moderation, and the power are also changing. These effects were interacting with the control rod drive mechanism, which was not modeled in this stand-alone simulation. Nevertheless, the core heating power is in strong correlation with neutronic power, and this was a measured quantity. Consequently, the thermal power (i.e. the heat source) had to be approximated with a table in the input, as shown in Figure 11.

Another important parameter was the turbine control valve (TCV) position. The opening characteristics of the valve strongly affected the turbine steam demand. Since the turbine was modeled with a boundary condition, approximation of the TCV opening curve was reconstructed "manually" from the steam line pressure and flowrate data (Figure 12).

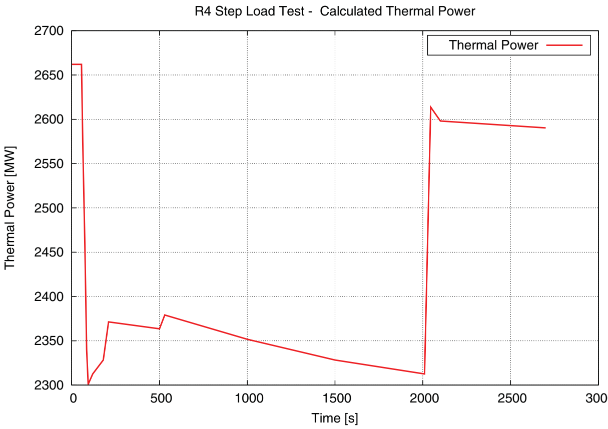

Figure 11 Approximation of thermal power

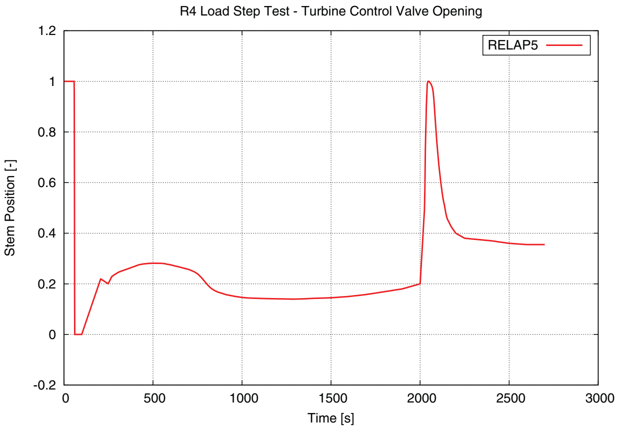

Figure 12 TCV valve opening

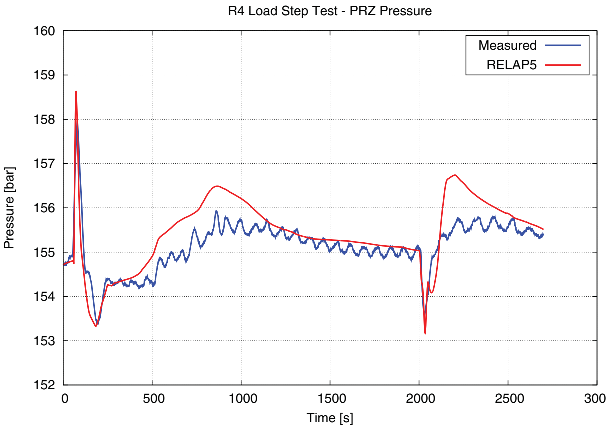

Figure 13 Pressure in the pressurizer

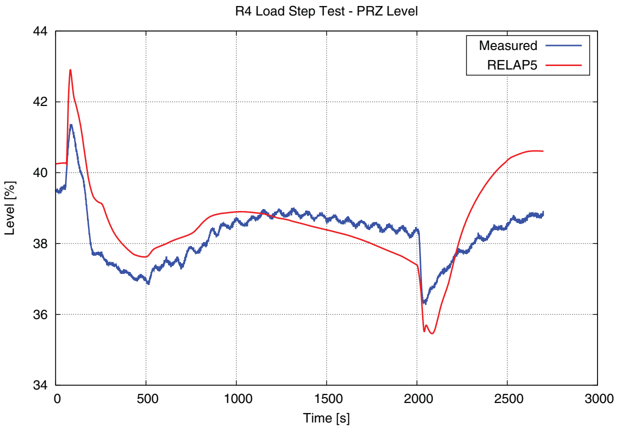

Figure 14 Level in the pressurizer

As it can be seen on Figure 13, re-pressurization of the system was observed immediately at the beginning of the transient. Timing and magnitude of the large initial pressure peak and the negative peak were captured very well by the code. The following slow pressure increase was slightly overestimated, reaching a maximum around 156.5 bar. The power was increased back again at approximately 2000 s. A valley of the pressure was also well represented in the code calculation. Behavior of the PRZ level is somewhat similar to the pressure data. The plot of the collapsed level in the PRZ is depicted on Figure 14. The initial level peak is slightly overestimated by less than 2 %. In absolute terms of the PRZ total level span of 10.303 m, a discrepancy of 2 % means approx. 0.2 m.

Concerning the narrow range level in the SG (Figure 15), performance of the RELAP5 model is excellent. The main qualitative difference is that the calculated values do not show the same oscillatory nature as of the measurements. It is due to the fact that the injected amount of feedwater is controlled in a simplified manner, with opening and closing the feed control valve in the model. However, it is somewhat more complex in reality: the velocity of the feed pump is coupled with the control valve with a hysteresis, which is reflected in the oscillation. Matching of the calculated and measured quantities is true also for the two main components that determine the SG level: the feedwater flowrate (Figure 16), and the steam flowrate (Figure 17). These parameters are also well simulated by the code.

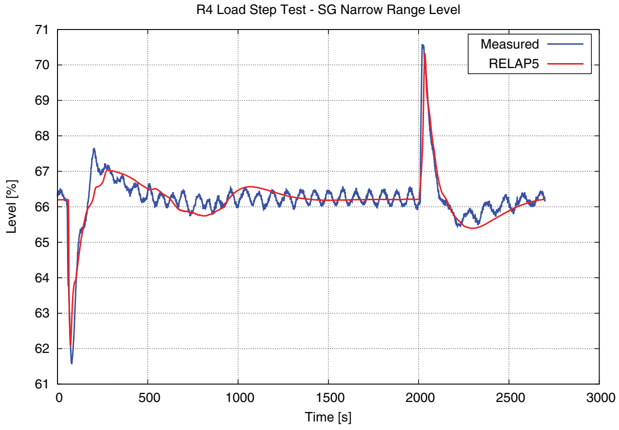

Figure 15 Level in the steam generator

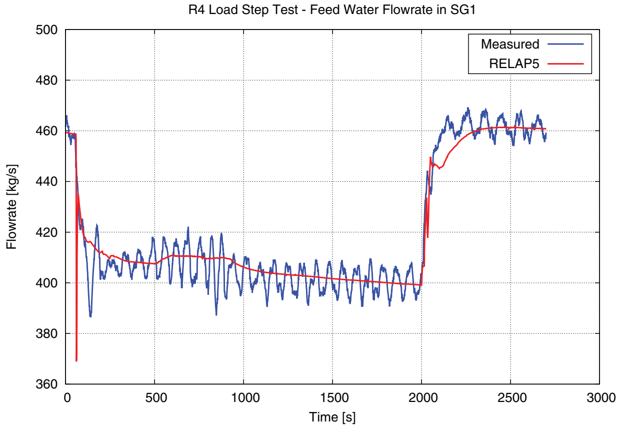

Figure 16 Feed water flowrate in SG1

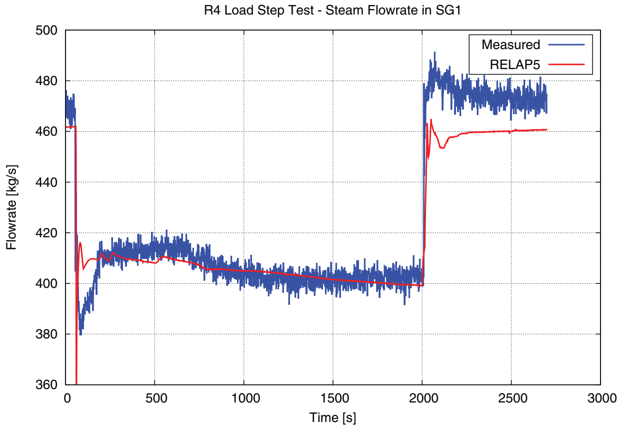

Figure 17 Steam flowrate from SG1

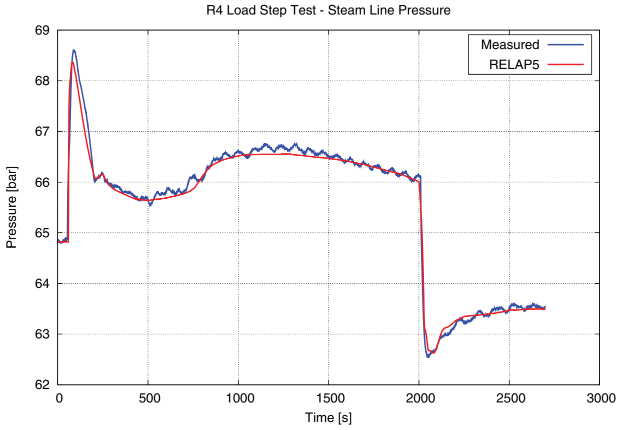

Figure 18 Steam line pressure

The loop fluid temperatures in the hot leg (Figure 19) and in the cold leg (Figure 20) are very important parameters. These quantities are closely correlated with the energy transfer in the reactor. Specifically, the temperature change over the reactor core (ΔT) and the average primary temperature (Figure 21) describe in the best way, how the thermal power is transferred to the SG secondary side.

During the process of model building and testing, it frequently happens that a certain plant model is able to predict the core ΔT adequately but it fails to match the average temperature, or vice versa. In such a situation, the flowrates or other contributing factors of energy transfer are not simulated properly. This is not the case for the current Ringhals 4 model presented here. As it is demonstrated, the general trends and quantities are sufficiently well reproduced by the code.

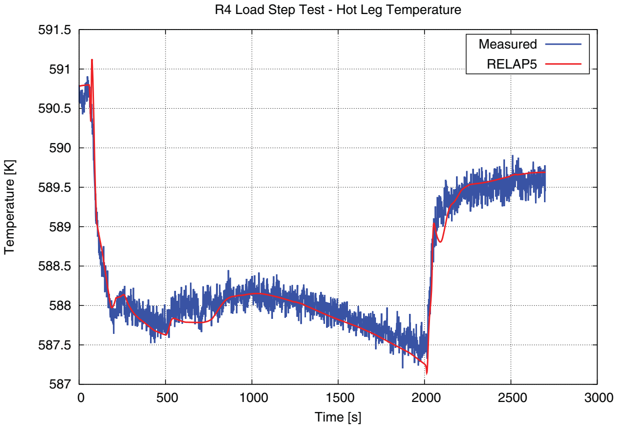

Figure 19 Temperature in the hot leg

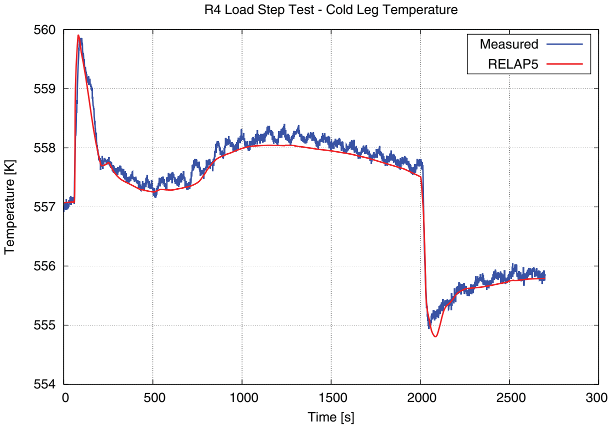

Figure 20 Temperature in the cold leg

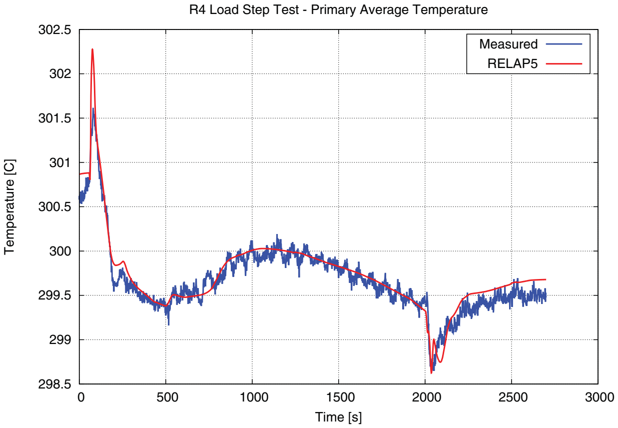

Figure 21 Average temperature in the primary

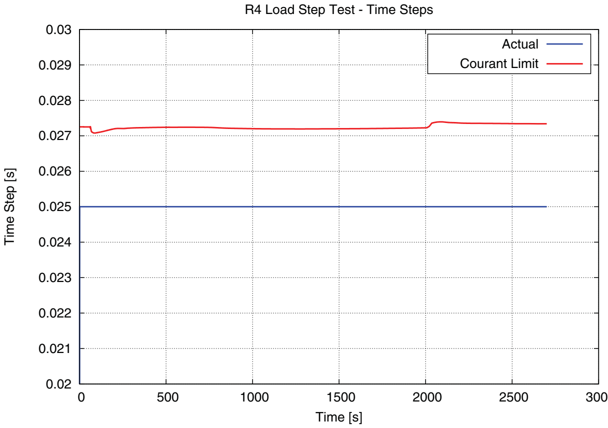

Figure 22 Time steps in the calculation

Local peaks are present both in the measured values and the simulated parameters at approx. 200 s and 500 s (Figure 19 and Figure 21). These are consequences of the power changes, shown in Figure 11.

A user-requested time-step size of 0.025 s resulted in a stable running of the simulation. Except the first initial steps, this time-step was adequate for an economical running of the code, very close to the Courant-limit but never exceeding that (Figure 22).

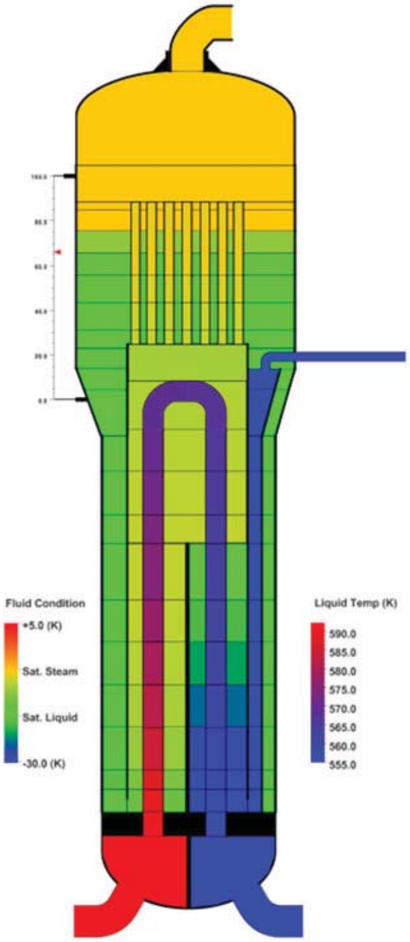

Figure 23 Fluid conditions in the SG

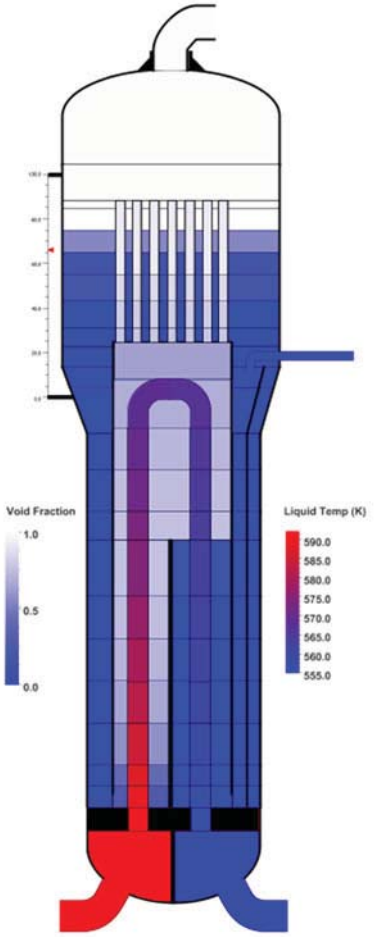

Figure 24 Void fractions in the SG

The RELAP5 package [5] originating from the US NRC is extended with a graphical front-end, called SNAP [6]. With the help of this tool, the user can animate various parameters of any performed calculations. Figure 23 and Figure 24 are examples for the animation masks that can be very helpful for visualization of parameter distributions (e.g. fluid conditions or void fractions) within the component models.

5. CONCLUSIONS

An important stage of the R4 model development process has been summarized in the current study. Incorporation of the new components into the full plant model was a challenging task because it was the first opportunity to validate the new model against a startup test, without having prior experiences with the prototype geometries, such as the AREVA design SGs or the PRZ.

On the basis of the presented steady-state and transient results, it can be concluded that the over-all performance of the R4 RELAP5 model is good. The general trends during both parts of the transient (i.e. the power decrease and the power increase), were well predicted by the code. Quantitative matching of the parameters was satisfactory, which resulted in a successful dynamical validation of the model. The primary to secondary side energy transfer was sufficiently well simulated. The R4 model is suitable for analysis of other types of transients, already in its current state of development.

However, as always, there are possibilities for improvement. For instance, further information has to be obtained about the characteristics of the turbine control valve because this is a crucial parameter strongly influencing the behavior of the secondary side. During the first evaluations of the PRZ related control system, minor discrepancies were observed between the measured and the calculated pressures. Therefore, further refinement of the relevant component model will be carried out, preferably by using new test data as soon as those will be available. Also, there is an initiative for extension of the thermal-hydraulic model with neutron kinetic simulation, in order to perform coupled calculations in the future.

ACKNOWLEDGMENTS

Chalmers University is very thankful for the financial support provided by the Swedish Radiation Safety Authority (SSM) for the current model validation project. The authors would like to thank the plant personnel at Ringhals AB for their efforts with recording and providing the test data for code validation.

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