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RELAP5 to TRACE model conversion for a Pressurized Water Reactor

Master's thesis

Federico López-Cerón Nieto

Department of Physics
Division of Subatomic and Plasma Physics
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Chalmers University of Technology
SE-412 96 Gothenburg
Sweden
E-mail: lpezcern@student.chalmers.se

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Abstract

Safety is an important concern in Nuclear Power Plants because of large radioactive inventory produced during operations. Therefore, an accurate analysis of the system under normal and accidental conditions is necessary and relies on the use of complex computational capabilities. In particular, thermal-hydraulic system codes such as RELAP5 and TRACE can be applied to study the cooling of a reactor core under a variety of possible scenarios (e.g., in the case of a loss of coolant accident).

The focus of the current thesis is on the development of a TRACE model for a Pressurized Water Reactor. Such a model is derived from an existing RELAP5 model that was built at the Universitat Politecnica de Valencia (UPV). The procedure consists of an automatic conversion by applying the software SNAP. Following the automatic conversion, an extensive post-processing is needed to correct possible bugs and improve the new model. In addition, a second TRACE input deck is created, where a special component available in TRACE is used for the reactor core region, namely the VESSEL component.

Different types of calculations are performed with the two new TRACE models and with the RELAP5 model, so that the correctness of the TRACE models can be verified. First, steady-state and transient simulations using only TRACE and RELAP5, are run. Then, since in Pressurized Water Reactors there is a strong coupling between the coolant conditions and the reactor power, calculations based on the coupling of TRACE and RELAP5 to the 3-D neutronic core simulator PARCS, were also considered.

The analysis shows that the RELAP5 model and the TRACE model without the VESSEL component lead to similar results. On the other hand, the TRACE model with the VESSEL component, which consists of only a single channel for the reactor core region, requires further refinements.

Acknowledgments

I would first like to sincerely thank my two thesis supervisors Agustín Abarca of the Universitat Politecnica de Valencia (UPV) and Paolo Vinai of Chalmers University of Technology.

Certainly, this work would have been finished without Agustín´s support and help. He has always been willing to bear a hand; the last weeks of work were especially demanding for both of us and he always encouraged me into not losing patience. A long share of this work belongs to him.

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I would also like to acknowledge my appreciation to all the members in the department of Nuclear Engineering at UPV, for giving me the possibility of carry out this work with them. In particular, Rafa Miró and Gumersindo Verdú, for the opportunity to have the first look into the Spanish nuclear scene.

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1 INTRODUCTION

1.1 Nuclear Energy

Nuclear energy is one of options for generating electricity with reduced greenhouses gas emissions, and it currently covers a share of 10.6% of the worldwide electricity production [1]. There are 450 nuclear power reactors in operation with a total net installed capacity of 392082 MW_e, with 2 nuclear power reactors in long-term shutdown and 60 nuclear power reactors under construction [2].

In Nuclear Power Plants (NPPs) energy is provided from a controlled nuclear fission chain reaction, where neutrons are used to bombard and split fissile nuclides contained in the nuclear fuel. The nuclear fuel is arranged in the reactor core and the fissile material may be Uranium-235, Uranium-233, Plutonium-239 and Plutonium-241.

Figure 1 shows how a fission reaction starts. The energy is released in the form of kinetic energy of the fission products (here stated as $^{141}_{56}Ba$ and $^{92}_{36}Kr$, although there is a huge combination of fission products), high-energy (fast) neutrons and gamma radiation.

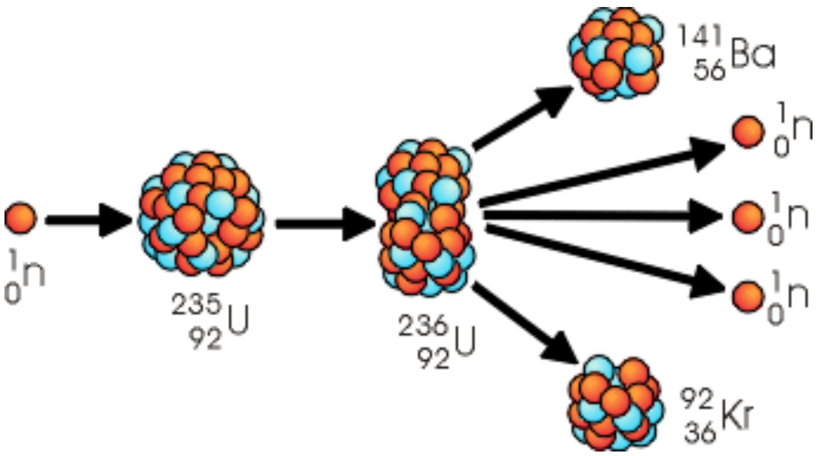

Figure 1 – Fission reaction.

For each fission reaction, 2-3 high-energy neutrons become available, so that new fissions are possible and a self-sustaining chain reaction can be achieved. The energy is then transferred to a fluid, which drives a turbine connected to an electrical generator.

Nowadays, the most common commercial nuclear reactors are Light Water Reactors (LWRs), which use light water (H_2O) as coolant and neutron moderator.

Nuclear fuel is manufactured from natural uranium that has a small fraction of the fissile isotope Uranium-235 (about 0.7%). However, a higher fraction of fissile material is needed to operate feasibly a nuclear reactor; therefore, enrichment in terms of Uranium-235 or addition of Plutonium-239 and 241 is applied. In LWRs the nuclear fuel consists of uranium oxide (UOX) with low-enriched uranium (3-5% of U-235), or mixed oxide (MOX) with a low content of plutonium (2-11%) combined with natural or depleted uranium.

The moderation process is essential to maintain a chain reaction in this type of reactors. Since, in general, the probability of fission to occur increases when the energy of the neutron decreases, the fast neutrons released from the fissions have to be slowed down from 14 MeV to the thermal energies (~0.025 eV) by the moderator, in order to have a good probability of giving new fissions within such nuclear fuels.

Since water plays the double role of coolant and moderator, there is a strong coupling between thermal-hydraulics and reactor power in LWRs: any variation of the coolant/moderator conditions may affect the neutron population (thus the number of fissions and the energy released in the process), and vice versa.

Two main general designs are available for LWRs, namely Boiling Water Reactor (BWR) and Pressurised Water Reactors (PWR).

1.1.1 Boiling Water Reactors

A BWR is such that water boiling occurs in the core, and the vapour goes directly from the reactor core to the turbine (Figure 2). There is a separator in the upper part of the vessel that reduces the humidity content of the coolant so that the blades of the turbines will not be damaged. After the turbines, the exhausted vapour is then condensed and sent back to the core. The operative pressure of this system is about 68 bar.

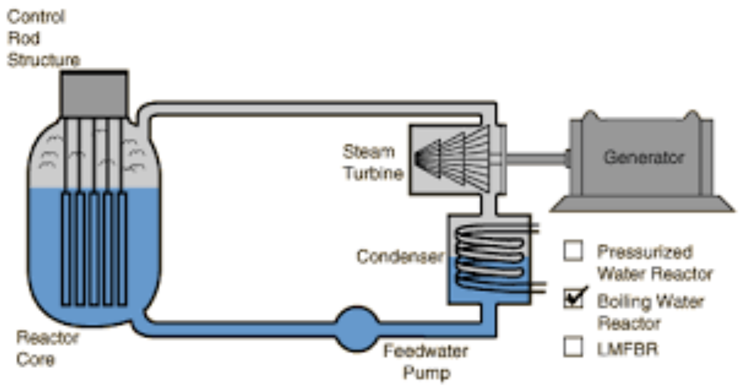

Figure 2 – Schematic of a Boiling Water Reactor (BWR).

In BWR, the power supplied by the fuel is controlled by two means:

I. **Control rods:** They consist of material with high neutron absorption cross-sections, so the neutrons released by the fission can be absorbed in the control rods, and will not give fissions. They are used for the start-up of the reactor, for controlling the power during operations, and for quick reactor shutdown.
II. **Main circulation pumps**: They are used when operating at high power (above 70%). By varying the mass flow through the main circulation pumps, the amount of coolant/moderator in the core changes and so the power of the reactor will vary as well. For instance, when the mass flow is decreased by the main circulation pumps, less amount of moderator will be present in the core

and fewer neutrons will be slowed down to a suitable energy range for fission and the power, therefore, will decrease.

III. **Boron**. In case of emergency, boron, which is a strong neutron absorber, can be added in the coolant.

In this type of reactors, each fuel assembly consist of 74-100 fuel rods, with around 800 fuel assemblies in the reactor core, summing up around 140 tons of enriched Uranium [3].

1.1.2 Pressurizer Water Reactors

A PWR includes a primary and a secondary circuit (Figure 3). The primary coolant flow removes the energy released by the fissions from the core, and goes to a heat exchanger, where vapour for the turbine is produced in the secondary side. The primary coolant is kept liquid by imposing a relatively high pressure in the primary loop (about 155 bar).

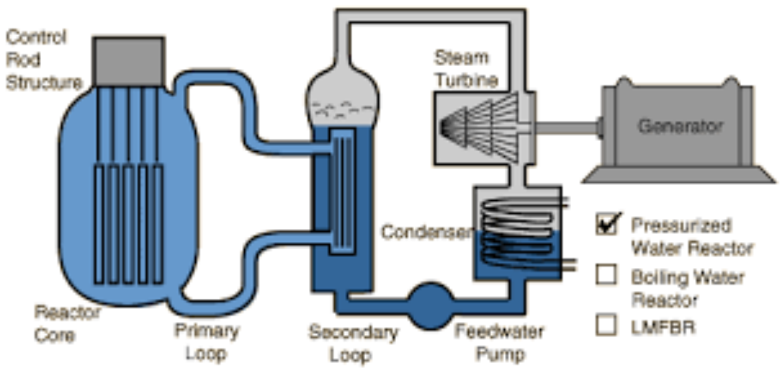

Figure 3 - Schematic of a Pressurized Water Reactor (PWR).

In PWR the control of the core is mainly made by:

I. **Control Rods**: Mainly used during the start-up and shutdown operations of the reactor.

II. **Boric acid injection**: This is the main way of controlling a PWR. It consists of the variation of the concentration of the boric acid in the primary system of the reactor. Boron absorbs neutrons, so its injection in the core will decrease the population of neutrons and therefore the power.

PWR usually have a 17x17 fuel rods array in one assembly and around 200 assemblies. In total there is approximately 86 tons of enriched Uranium on it [3].

The pressure in the primary system is controlled with a pressurizer, which is a vessel placed in one of the refrigeration loops in the primary side that is partially filled with water, being heated to the saturation temperature of the desired pressure by electrical heaters.

The turbines do not share the same circuit than the water flowing through the core and therefore it does not need to be shielded.

As drawbacks, the high pressure requires an extra cost in terms of equipment of piping and vessel. Boric acid is corrosive to carbon steel and may cause radioactive corrosion products. However, filters can be added to the system in order to remove those corrosion products.

1.2 Safety Analysis of Nuclear Power Plants

During the operation of a NPP, fission products and actinides are generated from the nuclear reactions. Such nuclei are unstable and decay, emitting radiations. Fission fragments mainly decay with beta emissions together with gamma emissions. Actinides are due to neutron capture in uranium-238 (which is about 99.2% of natural uranium and 96.5% of the fuel) and in other heavy isotopes that can be found in the nuclear fuel (e.g., thorium-232), and they are alpha emitter, with medium and long half-lives.
In view of this, a large amount of radioactivity is accumulated in the reactor core, and represents a severe hazard to workers, public health and the environment. Such a radioactive materials should be safely confined in the plant so that risk for any contamination is minimized. This objective is achieved by implementing several physical barriers between the radioactive isotopes and the surroundings, e.g. the fuel cladding, the reactor vessel and the pipes of the primary coolant loop, the reactor containment. Moreover, control and safety systems are included in order to protect the physical barriers when abnormal situations occur. A typical safety system is, for example, the Emergency Core Cooling System (ECCS) that can provide coolant in the case of a break in the primary loop.

In the demonstration of the safety of a NPP, simulations of normal operations, anticipated transients, and accidents, play an important role. Therefore, computer codes have been extensively developed to analyse the different aspects of a nuclear reactor. For instance, there are codes for the evaluation of the fuel behaviour, reactor kinetics, thermal-hydraulics, radiological protection, radionuclide transport, probabilistic risk assessment, etc.

Thermo-hydraulic system codes are used to study the physical response of the reactor system to postulated transient and accidental events and determine if any failure may lead to a significant release of radioactivity. Two codes of this type are RELAP5 and TRACE, both developed with the support of the United States Nuclear Regulation Commission (US-NRC).

Neutron kinetics codes are applied to calculate the neutron flux and power distribution in the core, under both steady-state and transient conditions. An example is PARCS, which is supported by US-NRC.

The analysis of transient scenarios in which there is a strong coupling between coolant/moderator conditions and reactor power distribution, is usually performed by coupling a system code with a 3-D core neutronic code.

1.3 Objective and Structure of the Thesis

The current project aims to build a TRACE input model for the primary coolant system of a PWR. For this purpose, an existing RELAP5 input model is employed as starting point. Such a model can be converted into an equivalent TRACE model, with the software package SNAP. The outcome of this automatic procedure however needs a significant post-processing in terms of corrections, refinements and improvements. In order to verify the correctness of the new model, different types of calculations are performed with the RELAP5 and TRACE models, and are compared.

The thesis is structured as follows. In chapter 2, the codes RELAP5, TRACE and PARCS that are used in the work, are introduced. In chapter 3 the existing RELAP5 input deck of a PWR is first described; then the procedure for the conversion of the model from RELAP5 to TRACE, and the specific modifications required in the post-processing phase, are explained; finally a second, alternative TRACE input model is derived by making use of a special component, namely the VESSEL component. In chapter 4 the results from the comparison between the RELAP5 and TRACE calculations are presented and discussed. In chapter 5 conclusions are drawn.

2 Computer Codes

In this chapter the computer codes used in the project are introduced. In particular, the thermal-hydraulic system codes RELAP5 and TRACE are discussed in terms of their basic, theoretical principles, and their modeling philosophy. The neutronic core simulator PARCS is also presented.

2.1 RELAP5

RELAP5 [4,5] is a thermal-hydraulic system code that is applied to simulate the reactor system behavior under operational and accidental conditions. In particular it is used for the analysis of transients in LWRs, e.g. Loss-of-Coolant Accidents (LOCAs), Loss-of-Flow Accidents (LOFA), Station Black-Out (SBO).
The code is based on time-dependent, one-dimensional field equations that model the coolant two-phase flow, and the presence of Boron and non-condensable gases [4].

2.1.1 Conservation equations for the coolant flow

In LWRs boiling of the coolant is a key phenomenon; so modeling capabilities for the simulation of liquid/vapour two-phase flow is necessary. In RELAP5 separate mass, momentum and energy conservation equations are specified for the liquid and vapour phases. The independent variables are time (t) and space (x); the dependent variables are pressure (P), internal specific energies of vapour and liquid (U_g, U_f), vapour volume fraction (α_v), velocities of vapour and liquid (v_g, v_f).
The mass conservation equation for the generic phase k is:

$$\frac{\partial}{\partial t}(\alpha_k \rho_k) + \frac{1}{A}\frac{\partial}{\partial x}(\alpha_k \rho_k v_k A) = \Gamma_k \tag{2.1}$$

where α_k is the volume fraction, ρ_k is the density, A is the flow area, and v_k is the velocity.

The first term on the left hand side computes the change in mass of the system due to the variation of density with the time and the second term the change in mass of the system due the fluid entering and leaving the system. The right hand side of the equation is the volumetric mass exchange rate. No mass sources or sinks is considered, so the generation of one phase comes from the disappearance of the other phase, according to processes of liquid vaporizing and vapour condensation. Then the source term can be simplified as follows:

$$\Gamma_l = -\Gamma_g \tag{2.2}$$

The momentum conservation equation for the phase k (being the other phase denoted as r) is:

$$
\begin{aligned}
&\alpha_k \rho_k A \frac{\partial v_k}{\partial t} + \frac{1}{2} \alpha_k \rho_k A \frac{\partial v_k^2}{\partial x} \\
&= -\alpha_k A \frac{\partial P}{\partial x} + \alpha_k \rho_k B_x A - (\alpha_k \rho_k A) FW_k v_k + \Gamma_k A (v_{ki} - v_k) \\
&- (v_k \rho_k A) FI_k (v_k - v_r) - C \alpha_k \alpha_r \rho_m A [\frac{\partial (v_k - v_r)}{\partial t} + v_r \frac{\partial v_k}{\partial x} \\
&- \frac{v_k \partial v_r}{\partial x}]
\end{aligned} \tag{2.3}
$$

The left-hand side includes the contributions due to the variation of velocity with respect to time and the variation of the square of the velocity with respect to space. The right hand side of Eq. (2.3) is the contribution of the gradient of pressure P, body force (gravity and pump head), wall friction, momentum transfer due to interface mass transfer, interface frictional drag and force due to virtual mass. The term FW_k accounts for the wall frictional drag and the term FI_k is part of the interface frictional drag. The energy conservation equation for the phase k is:

$$
\begin{aligned}
&\frac{\partial}{\partial t} (\alpha_k \rho_k U_k) + \frac{1}{A} \frac{\partial}{\partial t} (\alpha_k \rho_k U_k v_k A) \\
&= -P \frac{\partial \alpha_k}{\partial t} - \frac{P}{A} \frac{\partial}{\partial x} (\alpha_k v_k A) + Q_{wk} + Q_{ik} + \Gamma_{ig} h_k^* + \Gamma_w h_k' + DISS_k
\end{aligned} \tag{2.4}
$$

where Q_{wk} are the phasic wall heat transfer rate per unit volume, Q_{ik} are the phasic interfacial heat transfer rate per unit volume, h_k^* are the phasic enthalpies associated with bulk interface mass transfer, h_k' are the phasisc enthalpies associated with Wall interface mass transfer and DISS is the energy dissipation function.

The conservation equations are complemented by closure relationships in order to determine mass, momentum and energy transfer between the phases and between the phases and the wall. Examples of these relationships are the correlations for interfacial friction, fluid-wall friction, wall-to-fluid heat transfer, condensation, etc. etc.

2.1.2 Boron Transport

Boron is a neutron absorber and its injection in the coolant is one of the options used to control the nuclear reactions (so the power) in the reactor core. Its transport is modeled with several assumptions: neither energy nor inertia will be carried by the solute; and the solute will be transported in the liquid phase. Then the equation can be written as:

$$
\frac{\partial \rho_B}{\partial t} + \frac{1}{A} \frac{\partial \left(\rho_B v_f A \right)}{\partial x} = 0 \tag{2.5}
$$

Where ρ_B is the density of Boron.

2.1.3 Non-condensable Gases

Non-condensable Gases, such as nitrogen, hydrogen and air, can come into play under several scenarios that are relevant to reactor safety analysis. They are normally present as part of the vapour phase. In RELAP5 it is assumed that the non-condensable gas has the same temperature as its surroundings, it does not affect the properties of the vapour and it has the same velocity than the vapour phase.

Due to the assumptions mentioned above, there is no need to include them in the momentum and energy conservation equations, and only an extra mass conservation equation needs to be added:

$$\frac{\delta(\alpha_g \rho_g X_n)}{\delta t} + \frac{1}{A}\frac{\delta\left(\alpha_g \rho_g X_n v_g A\right)}{\delta x} = 0 \tag{2.6}$$

2.2 TRACE

TRACE (TRAC/RELAP Advanced Computational Engine) [6,7,8] was developed from the combination of four main codes, namely TRAC-P, TRAC-B, RELAP5 and RAMONA.
The code is used to perform best estimate analysis of LOCAs, and other common operational and accidental scenarios in LWRs. It includes several capabilities, such as: multidimensional two-phase flow, non-equilibrium thermo-hydrodynamics, generalised heat transfer, reflood, level tracking and reactor kinetics.
The field equations [6] are similar to the RELAP5 ones, and a similar notation is also used.

2.2.1 Conservation equations for the coolant flow

The mass conservation equations for the liquid and vapor phases are [6]:

$$\frac{\partial\left((1-\alpha_g)\rho_l\right)}{\partial t} + \nabla\left[\left(1-\alpha_g\right)\rho_l \overline{v_l}\right] = -\Gamma_g \tag{2.7}$$

$$\frac{\partial\left((\alpha_g)\rho_g\right)}{\partial t} + \nabla\left[\left(1-\alpha_g\right)\rho_g \overline{v_g}\right] = \Gamma_g \tag{2.8}$$

The momentum equations for the two phases are:

$$\frac{\partial\left(\left(1-\alpha_g\right)\overline{\rho_l}\overline{\vec{v_l}}\right)}{\partial t} + \nabla\left[\left(1-\alpha_g\right)\overline{\rho_l}\overline{\vec{v_l}\vec{v_l}}\right] = \nabla\left[\left(1-\alpha_g\right)\overline{T_l}\right] + (1-\alpha)\overline{\rho_l}\vec{g} - \overline{\vec{M_l}} \tag{2.9}$$

$$\frac{\partial\left(\left(\alpha_g\right)\overline{\rho_g}\overline{\vec{v_g}}\right)}{\partial t} + \nabla\left[\left(\alpha_g\right)\overline{\rho_g}\overline{\vec{v_g}\vec{v_g}}\right] = \nabla\left[\left(\alpha_g\right)\overline{T_g}\right] + \left(\alpha_g\right)\overline{\rho_g}\vec{g} + \overline{\vec{M_l}} \tag{2.10}$$

In Eq. (2.9) and (2.10), T_g is the full stress tensor and M_i represents the total contribution of time averaged interface jump conditions to momentum.
The energy equations are:

$$\partial\left[\frac{(1-\alpha_g)\overline{\rho_l}\left(\overline{e_1+\frac{V_1^2}{2}}\right)}{\partial t}\right]+\nabla\left[(1-\alpha_g)\overline{\rho_l}\left(\overline{e_1+\frac{P}{\rho_l}+\frac{V_1^2}{2}}\right)\overline{\overline{\vec{V_l}}}\right]$$
$$=-\nabla\left[(1-\alpha_g)\overline{\overline{\vec{q_l'}}}\right]+\nabla\cdot\left[(1-\alpha_g)\left(\overline{T_l\cdot\vec{v_l}}\right)\right]+(1-\alpha_g)\overline{\rho_l}\overline{\overline{g\vec{v_l}}}-\overline{E_\iota}+\overline{q_{dl}} \tag{2.11}$$

$$\partial\left[\frac{(\alpha_g)\overline{\rho_g}\left(\overline{e_g+\frac{V_g^2}{2}}\right)}{\partial t}\right]+\nabla\left[(\alpha_g)\overline{\rho_g}\left(\overline{e_g+\frac{P}{\rho_g}+\frac{V_g^2}{2}}\right)\overline{\overline{\vec{V_g}}}\right]$$
$$=-\nabla\left[(\alpha_g)\overline{\overline{\vec{q_g'}}}\right]+\nabla\cdot\left[(\alpha_g)\left(\overline{T_g\cdot\vec{v_g}}\right)\right]+(\alpha_g)\overline{\rho_g}\overline{\overline{g\vec{v_g}}}+\overline{E_\iota}+\overline{q_{dg}} \tag{2.12}$$

Where E_i represents the total contribution of time averaged interface jump conditions to transfer of energy, q' is the conductive heat flux, q_d is the direct heating (i.e. radioactive decay) and T is the full stress tensor.

2.2.2 Boron Transport

The transport of Boron is modeled in a similar way to the RELAP5 case. The boron does not affect the hydrodynamics directly as in RELAP5. However, it may affect hydrodynamics indirectly through neutronic-reactivity feedback. The liquid-solute concentration equation implemented in TRACE is:

$$\frac{\delta\left[(1-\alpha_g)m\rho_l\right]}{\delta t}+\nabla\left[(1-\alpha_g)m\rho_l\vec{v_l}\right]=S_m \tag{2.13}$$

Where m is the solute concentration.

2.2.3 Non-condensable Gases

Normally only one non-condensable specie is modeled in TRACE, although the user can insert multiple ones, increasing the running time by using extra field equations. The non-condensable mixture is in thermal equilibrium with any steam present and moves with the same velocity. Mass conservation equation for non-condensable gasses is:

$$\frac{\delta(\alpha_g \rho_n)}{\delta t} + \nabla[\alpha_g \rho_n \overrightarrow{v_g}] = 0 \tag{2.14}$$

2.3 Modeling strategy in RELAP5 and TRACE

The modeling approach of RELAP5 and TRACE is such that the hydrodynamic system is discretized in a network of fluid control volumes connected by junctions, consistently with the real flow path. When the simulation is performed, the field equations are solved for each of the control volumes according to the specified flow path.

The discretization (or nodalization) of the system is defined by the code user and is provided in an input file whose format depends on the code specifications. Several components are defined in order to simplify the hydrodynamic modeling. Some components are useful to reduce the amount of information needed in the input file. For instance, the PIPE component that is available in both RELAP5 and TRACE, consists of a series of control volumes and junctions. By using the PIPE component, the user only has to insert the data for one control volume and the code will automatically set that value to all the control volumes making up the PIPE component. Other components allow including data for ad hoc features and additional processing. One example is the VALVE component where type, characteristics and data for determining the flow area of the valve can be entered. Specific components can be used to provide time-dependent temperature, pressure and velocity of the flow at the inlet and outlet of the system. In fact, the boundary conditions are given with the time-dependent volume and time-dependent junction components in RELAP5, and with the FILL and BREAK components in TRACE.

Also, the heat flow paths across solid boundaries from/to the hydrodynamic flow (e.g., the heat transfer from the nuclear fuel rods to the coolant flow in the core), are modeled using heat structure components coupled to the fluid control volumes. The heat structures are characterized by a computational mesh over which, proper heat transfer equations are solved, so that heat fluxes and temperature distributions can be calculated.

Figure 4 shows and example of a RELAP5 nodalization scheme created for the simulation of a real system (namely, the Achilles test facility).

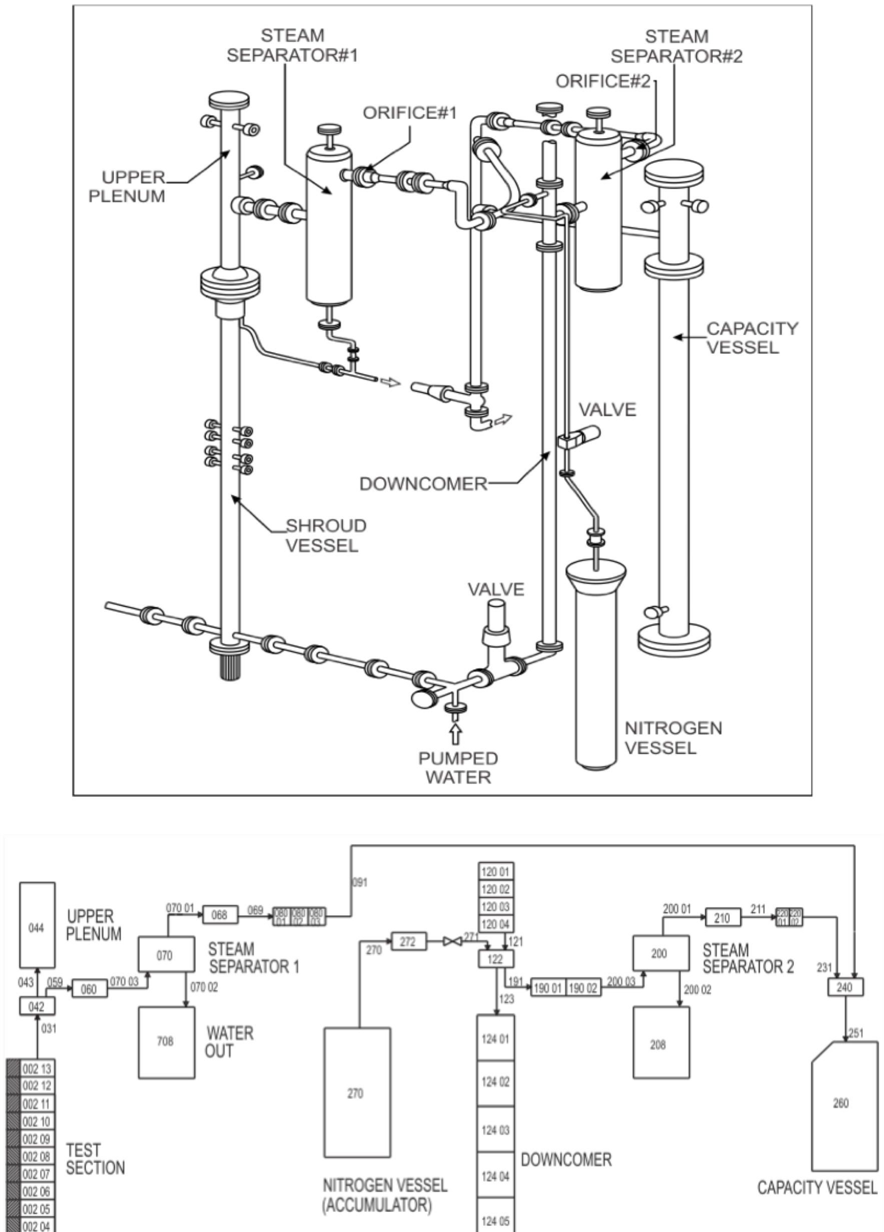

Figure 4 – Example of RELAP5 nodalization for a real facility (Achilles test facility). [9]

In nuclear power plants, a control and protection system is included. Such a system monitors crucial parameters of the plant; whenever an anomaly is detected, then the system automatically starts a sequence of actions in order to avoid negative consequences. For instance, when the control and protection system of a PWR measures a pressure in the primary coolant system higher than a prescribed value, then the relief valve of the pressurizer is automatically opened and the pressure is brought back to the normal level. Therefore it is important to consider the role of the control and protection system in the simulation of transient scenarios. In RELAP5 and TRACE it is possible to replicate the control logic of the plant by setting variables, trips and control blocks/components. The several output variables calculated from the code simulation for each time steps, can be served as control variables, to trips and control blocks. Trips are logical statements defined by the user, and they are either true or false. Control blocks/components allow the code user to design mathematical operations with the output variables to be performed during the simulation, so that the behavior of the control and protection system can be mimicked.

2.4 PARCS

PARCS is a time-dependent, 3-D core neutronics simulator that can be applied to compute the steady-state power distribution of the reactor core and the evolution of the power distribution under transient conditions. The time-dependent neutron flux is calculated by solving the following equations:

$$\frac{1}{V_g^m}\frac{d\phi_g^m}{dt} = \frac{1}{k_{eff}}\chi_{pg}\sum_{g=1}^{v} v_{pg}\Sigma_{fg}^m\phi_g^m + \chi_{dg}\sum_{k=1}^{\Lambda}\lambda_k C_k^m + \sum_{g'=1}^{v}\Sigma_{g'\ g}^m\phi_{g'}^m - \sum_{u=x,y,z}\frac{1}{h_u^m}(J_{gu}^{m+} - J_{gu}^{m-}) - \Sigma_{tg}^m\phi_g^m \tag{2.15}$$

$$\frac{dC_k^m}{dt} = \frac{1}{k_{eff}}\sum_{g=1}^{v} v_{dgk}\Sigma_{fg}^m\phi_g^m - \lambda_k C_k^m \tag{2.16}$$

where ϕ_g^m is the node-average flux, C_k^m is the precursor density , $J_{gw}^{m+}$is the surface average net current. Subscripts p and d stand for prompt and delayed neutrons and G and K are the numbers of neutron energy and delayed neutron precursor groups.

For transient calculations, PARCS is usually coupled to TRACE or RELAP5. In fact the macroscopic cross sections Σ that are needed in Eq. (2.15) and (2.16), depend on the coolant/moderator properties and the fuel temperature; on the other hand, the possible changes of the power distribution during the transient affect the thermal-hydraulic state of the core. In view of this, TRACE or RELAP5 are used to estimate the coolant properties and the fuel temperature with respect to the power distribution calculated by PARCS, at a certain time step of the transient calculation; the macroscopic cross-sections are updated with these coolant properties and fuel temperature; a new power distribution is computed accordingly, and it is available to the thermal-hydraulic code for the next step.

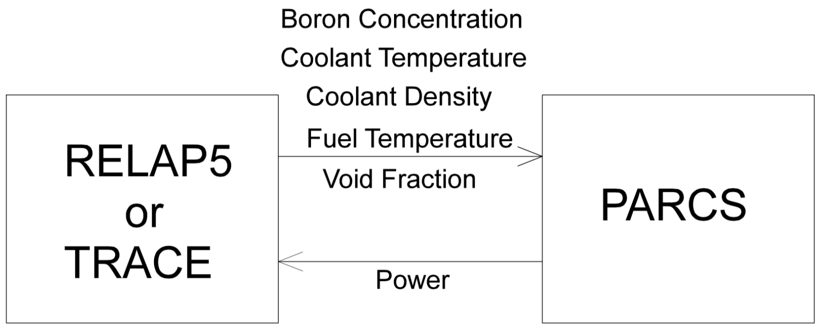

Figure 5 – Schematic of Thermo-hydraulics/Neutronics coupling.

3 Conversion of an input model of a PWR: from RELAP5 to TRACE

The starting point of the work is an existing RELAP5 input model of a three-loop PWR that was developed and applied at the Polytechnic University of Valencia. This input model is first introduced. The methodology for the conversion from RELAP5 into TRACE and the resulting TRACE input model are discussed. Finally the TRACE input deck was modified in such a way that the vessel region of the plant was described by the VESSEL component that is available in TRACE.

3.1 RELAP5 input model

The RELAP5 input model includes the reactor vessel with the core and the 3 primary coolant loops with the associated steam generators. As regards the secondary side, only the flow in the steam generators is considered along with the inlet and outlet conditions. In Figure 6, a simplified schematic of the nodalization is shown.

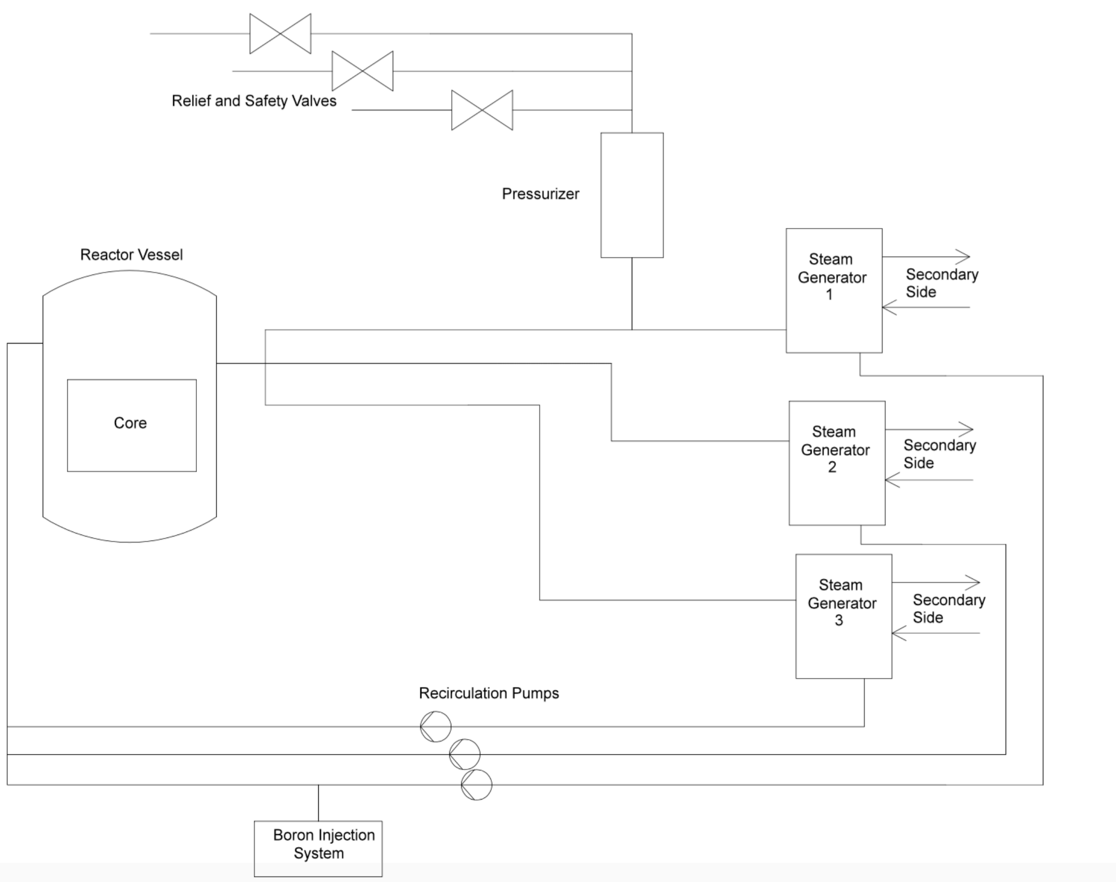

Figure 6 – Schematic of the primary side of the plant.

The details of the vessel region are given in Figure 7. The coolant flow from the steam generators enters the vessel. The main part of such a flow goes through the downcomer (modeled with an annulus component) to the lower plenum (described by a branch component), and reach the core region. The nodalization of the core region consists of 10 pipe components: 9 pipes are related to the flow in the fuel assemblies and 1 pipe is used for the core by-pass flow. Each of these pipes is coupled to a heat structure component in order to describe the heat transfer from the fuel elements to the fluid. All these heat structures share the same geometry but have different axial power shapes. The heat structures are geometrically modeled as a single fuel rod and have a surface multiplier so the contribution from all the physical fuel rods is properly counted. The coolant crosses the core from bottom to top, and exits the vessel from the upper plenum, which is discretized with a series of branch components.

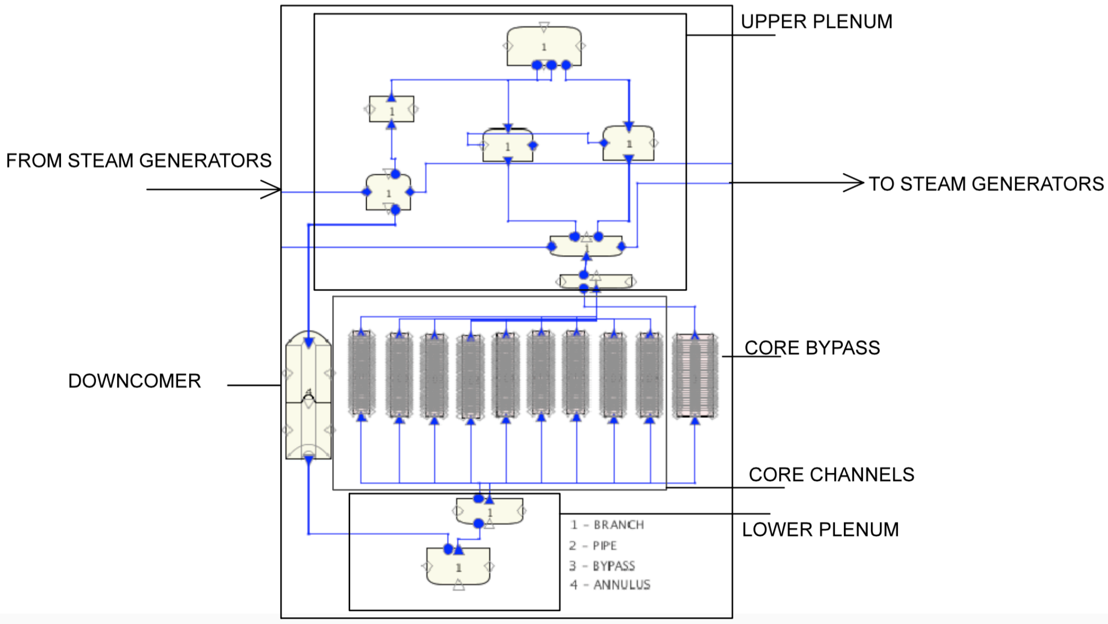

Figure 7 – Core/Vessel region in the RELAP5 model.

The heated coolant is sent, through the so-called hot leg, to the steam generators. The three steam generators are identical, and Figure 8 displays the nodalization for one of them. A pipe component is used for the flow on the primary side. The model of the secondary side consists of a branch component for the lower plenum, a pipe for the part that is heated by the primary flow, and a branch for the upper plenum. The boundary conditions are specified for the secondary side: at the inlet of the steam generator a time-dependent volume provides the pressure and temperature, and a time-dependent junction provides the mass flow rate; the outlet the pressure is given by a time-dependent volume. The heat transfer from the primary to the secondary side is reproduced with a heat structure component that couples the two pipes.
The primary coolant system includes two important systems: the pressurizer and the Boron injection system.

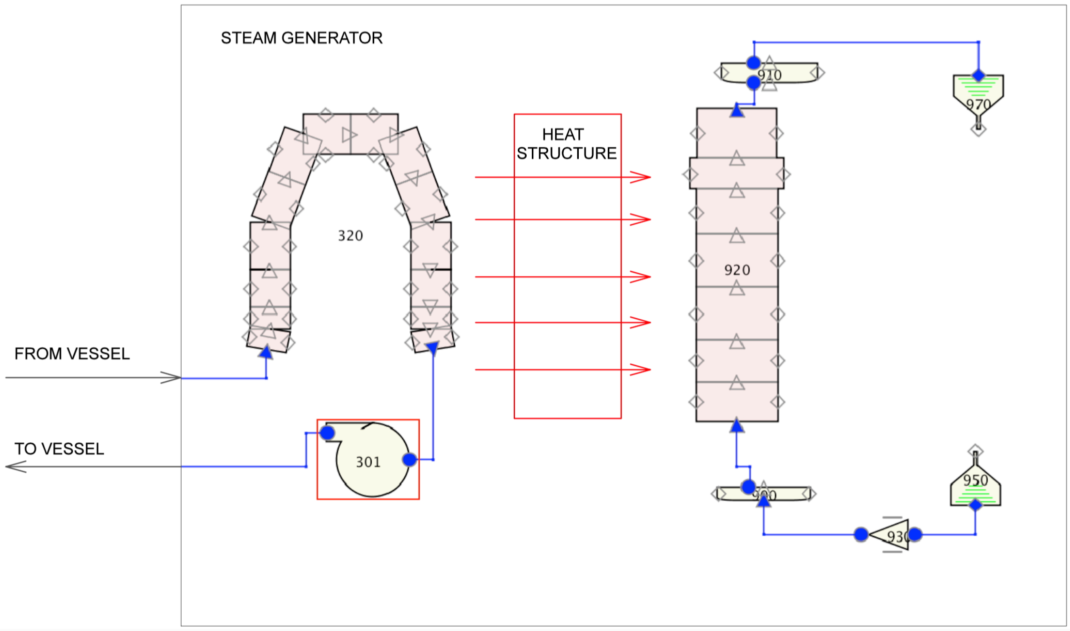

Figure 8 – Steam generator model in the RELAP5 model.

The pressurizer is needed to control the pressure of the system, and is connected to the hot leg of one of the three loops. The model (Figure 9) is built with a pipe for the pressurizer chamber; and branches for the connections between the inlet of the chamber and the loop, and between the outlet of the chamber and the lines with the relief and safety valves and with the valve that allows spraying cold water in the chamber. The behavior of the pressurizer is controlled by the control system. The upper part of the pressuriser has four different ramifications. Two of them are identical and have a time dependent volume, a pipe and a safety valve. There is another ramification that simply controls the pressure in the pressuriser with two valves. The last ramification is the spray valve. All valves component are controlled by the control system.

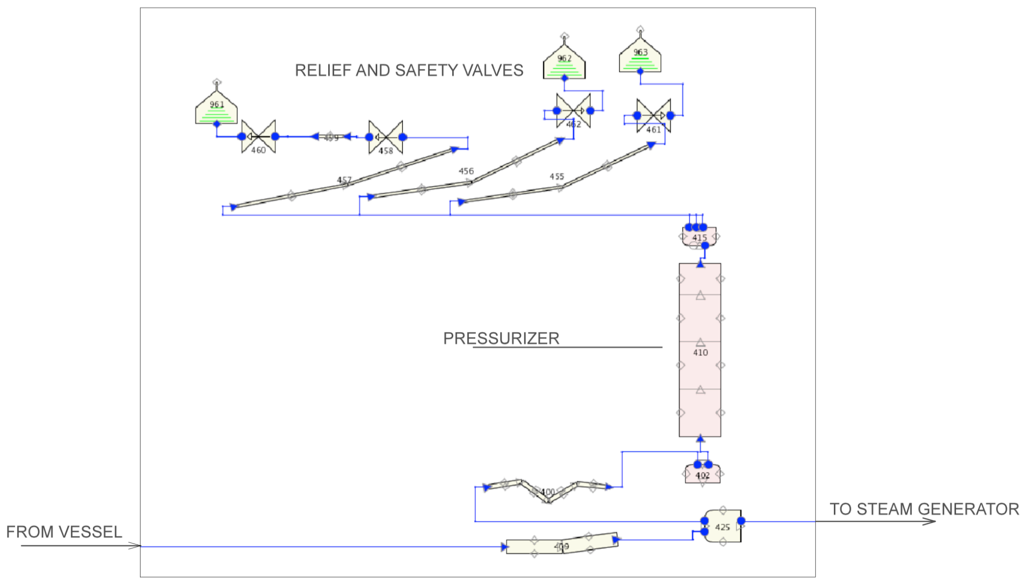

Figure 9 – Pressurizer system in the RELAP5 model.

The Boron injection system is used to control the core reactivity by adding Boron to the coolant in the cold leg of one of the loops. The nodalization scheme is shown in Figure 10. The flow comes from the steam generator and the most part of it is pumped to the vessel and small portion goes through the boron injection system. There are two time dependent junction and time dependent volume sets; one is for diluting down the concentration of boron and the other is for injection. Afterwards, the flow gathers in a pipe component to go to the vessel.

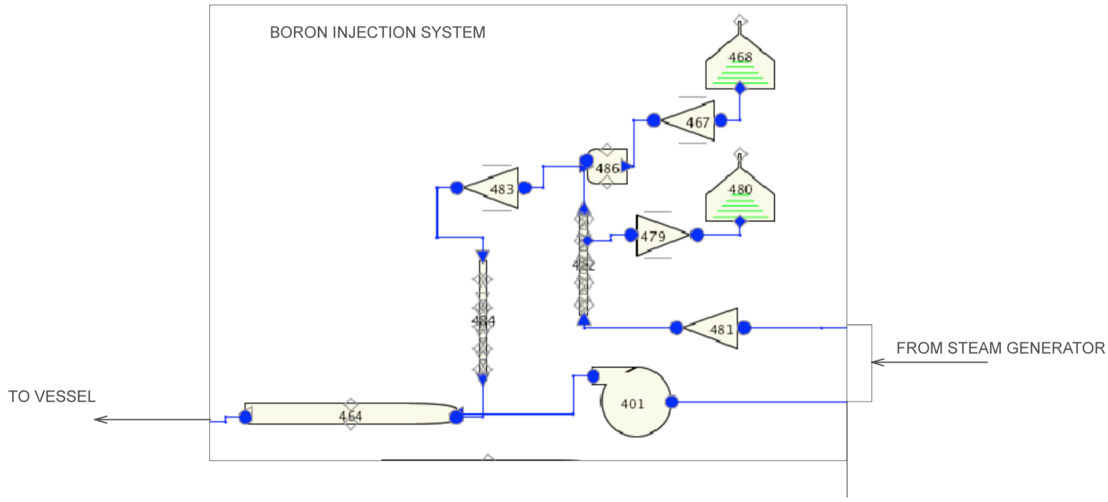

Figure 10 – Boron injection system in the RELAP5 model.

3.2 Procedure for the input file conversion

Methodologies for the conversion of a RELAP5 input model into a TRACE one, have been reported in previous works, e.g. [9]. Most of these works are based on a preliminary, automatic conversion performed with the software SNAP (Symbolic Nuclear Analysis Package), and a following refinement that is made by hand. In fact the automatic conversion can lead to a variety of errors due to the differences in the TRACE and RELAP5 specifications and capabilities, and to the particular modeling choices made in the model to be processed.
The procedure employed in the current project is similar to the ones used in the past, and it consists of these steps:

1) Convert the model by making use of the software SNAP; and fix the errors that were detected in the process.
2) Perform an input check of the new model with the TRACE code; and make the necessary corrections.
3) Perform a stand-alone, steady-state calculation with the TRACE model; compare the results against the ones obtained from the original RELAP5 model, for the same case; and make the necessary modifications of the TRACE model in order to fix the errors and the inconsistencies that have been found with respect to the RELAP5 case.

4) Perform the same transient calculations with the TRACE and RELAP5 models, coupled to the neutronic code PARCS; compare the two sets of results; and make the necessary modifications of the TRACE model in order to correct inconsistencies and errors.

3.3 Creating the TRACE input model

The application of the conversion procedure in the current project is discussed step by step, emphasizing the types of problems that were found and the corrective actions that were needed. Furthermore, the work for building the coupled TRACE/PARCS model is reported.

3.3.1 Automatic SNAP conversion (step 1)

The RELAP5 input deck is processed within SNAP, and a new TRACE input model is generated.
In the conversion, SNAP warns about errors in one pipe component of the loops and in the control system of the boron injection system. SNAP also finds that the flow area for some of the *PIPE* components must be stated. However, these parameters are automatically re-set during the TRACE calculation.

3.3.2 Input check with the TRACE code (step 2)

Once all the errors given by SNAP have been fixed, tests are run with the TRACE code, so that the new input file is processed and possible errors are identified.

One of these errors is related to the core region. The RELAP5 model of the fuel assemblies is such that each of the pipes used is coupled to a heat structure (see section 3.1). On the other hand, the TRACE model requires an additional component, namely a power component, for specifying the power of the heat structure. The SNAP conversion failed to create a correct power component, and adjustments were needed to have the right and complete set of input data for the initial power/temperature distribution in the heat structure. Moreover, different fuel assemblies may be characterized by different powers. In the SNAP conversion only one TRACE power component, with one initial axial power distribution, was generated for all the heat structures. Therefore additional power components, with different initial power distributions, had to be manually added.

Regarding the *PUMP* components, two aspects must be corrected. The trip controls of the pump components were not converted in the SNAP procedure, so they needed to be implemented in the TRACE input model. The second problem was the efficiency of the pump: information from the conversion was not sufficient for defining the parameter and missing data had to be entered in the new model (otherwise TRACE automatically set the efficiency equal to 85%).

In the TRACE input deck created by SNAP, no modeling of Boron is included; in fact the option 'solute tracking' is automatically set off. Therefore this parameter was changed throughout the entire TRACE input file, and the initial content of boron was added in all the hydraulic components of the primary coolant system. Control blocks for regulating the Boron concentration were also provided. The flag ISNOTB in the power component was checked in order to consider the boron effect on the reactivity.

Connections between components may be source of errors since the RELAP5 and TRACE approaches are somewhat different. For example, in RELAP5 a volume has numbered faces in such a way that components can be connected to any of these faces by specifying the relative number; in TRACE a volume has one main inlet and outlet junction, plus side junctions for which a proper angle is required. SNAP had difficulties in determining the TRACE angles from the RELAP5 faces. Also, hydraulic components may happen to be linked through the wrong junctions. Therefore angles of the side junctions and connections in the TRACE model, need to be carefully reviewed. One example of these kinds of issues concerned the inlet of the FILL component in the TRACE model of the boron injection system. After the conversion with SNAP, it was connected to the side junction of the adjacent PIPE component, and not to the inlet junction. Such an erroneous flow path led to a dead end, so that no boron could be delivered to the system.

Control systems had to be adjusted because of the differences between RELAP5 and TRACE. For instance, the liquid volume fraction is not directly available in TRACE calculations, so an additional control block was introduced for this variable.

3.3.3 Stand-alone, steady-state calculations (step 3)

The TRACE input model was corrected and code runs could be performed without fatal errors. The next step is to compare the TRACE stand-alone, steady-state calculations with the RELAP5 ones, check for possible discrepancies, and improve the new input file.

In this phase of the work, the main modifications involved: the wall roughness and the friction factors of most of the TRACE components; and hydraulic diameters and flow areas of those components that are connected to *BRANCH* components. In particular, the refinement of hydraulic diameters and flow areas were important in order to have correct cross-flows in the branch components used for the upper plenum of the vessel region.

The comparisons between these calculations are discussed in Chapter 4.

3.3.4 Coupling with the neutronic code PARCS (step 4)

TRACE and RELAP5 can be coupled to the neutronic code PARCS, by selecting specific options in the input files. For the coupling between RELAP5 and PARCS, the setting includes: EXT_TH=T and SYSNAME=RELAP in the PARCS input file. For the coupling between TRACE and PARCS, the options are such that: EXT_TH=T and SYSNAME=TRAC in the PARCS input deck; and ITDMR=1 in the TRACE input file.

The PARCS input file that was developed for the coupled calculation with RELAP5, is also kept in the coupled calculation with TRACE. Cross-sections needed for the neutronics are provided with external files. The values of the cross-sections are specified for three different Boron concentrations (i.e., 639, 1570 and 2500 ppm), with the control rods fully inserted and fully withdrawn. The code automatically interpolates between boron concentration groups and length of control rods insertion. The Xenon number density was also given as input data.

When coupling TRACE or RELAP to PARCS, an additional input file called MAPTAB, is prepared. This file contains the information about how the thermal-hydraulic mesh of the core is related to the neutronic one, i.e. which thermal-hydraulic channel of the core region (modeled in TRACE or RELAP5) is associated to which nuclear fuel assembly (modeled in PARCS). The two meshes are usually different. For instance, it is a common practice to use one thermal-hydraulic channel of the core for a group of fuel assemblies, so that the computational effort of the coupled calculation can be reasonable. On the other hand, neutronic core simulators require finer meshes and their burden on the calculation time is less limiting. Therefore, the MAPTAB file also includes weighting factors to take in to account the differences in meshes and to have a consistent coupling.
The MAPTAB file for the TRACE/PARCS calculation was derived from the one that was already available for the RELAP5/TRACE case.
The radial thermal-hydraulic map of the core is shown in Figure 11. Each number corresponds to a fuel assembly, and the fuel assemblies with the same number are assigned to the same TRACE pipe/heat structure. For the neutronic mesh, one radial node is used for one fuel assembly. As regards the axial nodalization of the core, both the TRACE pipes/heat structures and the neutronic model of the fuel assemblies consist of 34 nodes, in both of them the top and bottom axial levels correspond to the reflector and therefore they do not directly deliver any power.

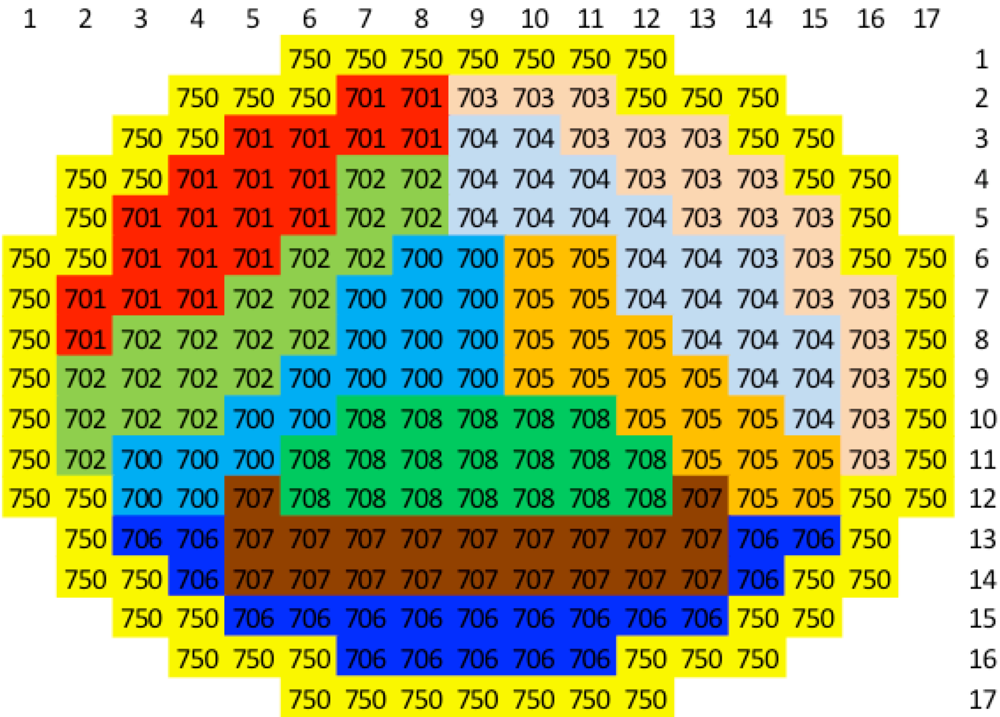

Figure 11 – Axial cross-section nodalization of the core.

The procedure used for running the coupled, transient calculation is as follows:

1) Perform a stand-alone, steady-state TRACE (or RELAP5) calculation.
2) From the converged TRACE (or RELAP5) solution of step 1), start the coupled, steady-state calculation.
3) From the converged coupled, steady-state solution of step 2), start the coupled, transient calculation.

The comparison between the results from the coupling between TRACE and PARCS, and the results from the coupling between RELAP5 and PARCS are discussed in Chapter 4.

3.4 TRACE input model with the VESSEL component

In the RELAP5 input deck the vessel region is modeled with a combination of different components (see Figure 7). As a result of the conversion, the TRACE nodalization of the vessel similarly consists of a network of components. However, the capabilities of TRACE allow the use of a single component for the multi-dimensional flow in this region, i.e. the so-called VESSEL component. Therefore, the TRACE input deck was modified with the VESSEL component, and a second, alternative TRACE input model of the plant is derived.

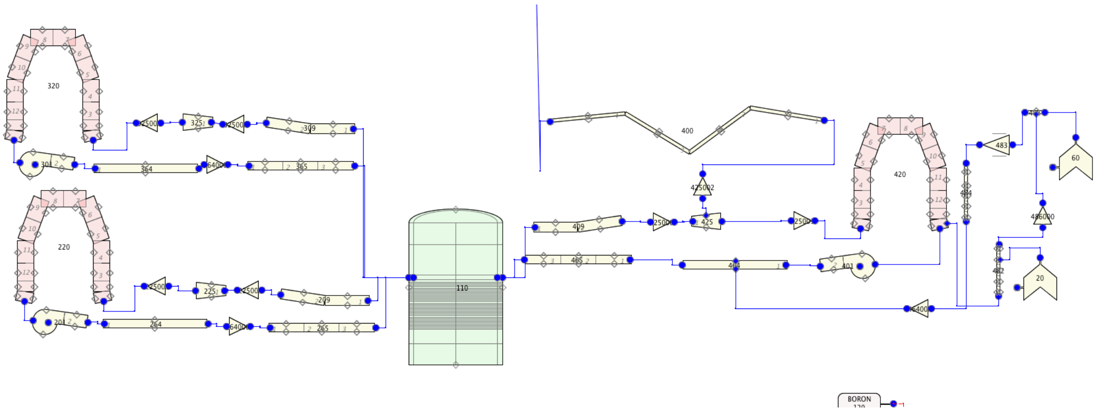

Figure 12 – TRACE nodalization scheme of the plant with the VESSEL component.

Figure 12 shows the TRACE nodalization scheme of the plant with the *VESSEL* component. The outlet and the inlet of the *VESSEL* component are connected with the three hot legs and the three cold legs, respectively. The three loops remain the same as in the first version of the input deck.
The *VESSEL* model includes 40 axial levels, 1 azimuthal sector and two radial rings, an external one for the downcomer and an internal one for the core region. Although the two input models with and without VESSEL component have equivalent geometry, some differences exist. Then some approximations were made in the radial and azimuthal flow areas as well as the hydraulic diameters.
When creating a *VESSEL* component, a HTSTR is required for each of the azimuthal sectors in order to model the heat losses from the vessel to the reactor containment. In addition, a HTSTR is also supposed to be defined for the shroud surrounding the core. The specifications of these heat structures were set at zero since the heat transfer from the vessel structures to the containment is not considered in the RELAP5 input file.
The signal variables that are used for the control systems and are related to the properties of the vessel region, were changed according to the new nodalization.

In the next chapter results calculated with this new model are also discussed.

4 RESULTS AND DISCUSSION

In this chapter comparisons between the RELAP5 input model and the two TRACE input models with and without the VESSEL component, are discussed. Three different simulation cases are considered: a stand-alone thermal-hydraulic calculation; a coupled thermal-hydraulic/neutronic calculation; a coupled transient calculation where the Boron content in the primary system of the plant is changed, affecting directly the reactor power.

4.1 Stand-alone, steady-state calculations

In this stage of the work, only stand-alone calculations with RELAP5 and TRACE models are considered.

The first comparison is based on a steady-state case, thus the simulations of the plant are performed under normal conditions. The most important parameters are checked in the vessel region, in the loops and in the pressuriser (see subsections 4.1.1 and 4.1.2).

The second comparison concerns a simplified transient simulation where a variation of Boron content in the cooling system is imposed without reactivity feedbacks. This study allows to check the correct implementation of the Boron Injection System (see subsection 4.1.3).

When analyzing these results, there are numerical oscillations at the beginning of the calculations. They are due to the initial conditions provided by the users that may be not fully consistent with the solution of the system of equations. These oscillations have no physical meaning, and disappear as soon as the simulation converges to a proper steady- state solution. In the current case, the quantities reach a steady-state condition between about 150 and 400 seconds.

4.1.1 Comparison between the RELAP5 model and the TRACE model without the VESSEL component

For the steady-state simulation, the discrepancies between the RELAP5 input model and the TRACE input model without VESSEL component are summarized in Table 1.

The numbering of the hydraulic components is the same than in the input file of RELAP5 or TRACE. Pipes modeling the fuel are numbered from 700 to 708 with the bypass being numbered as 750. The pipes modeling the fuel are linked to heat structures that provide them the heat corresponding to the fuel, adding a zero at the end of the hydraulic number gives the heat srtucture number. The hydraulic component in each loop that is placed after the pump in the direction of the flow has a X09 as hydraulic number, where X is 2 and 3 for the loops that are alike and 4 for the loop with the pressuriser and the boric injection system. The same applies for the pumps with an ending of 1 instead of 9. The hydraulic number 410 corresponds to the pipe modeling the pressurizer

Table 1 – Converged stand-alone, steady sate calculations: comparison between the RELAP5 input model and the TRACE input model without the VESSEL component.

RELATIVE DIFFERENCES FOR DIFFERENT PARAMETERS				
PARAMETER	**HYDRAULIC ID**	**TRACE**	**RELAP5**	**DEVIATION (%)**
MASS FLOW THROUGH CHANNELS MODELING THE FUEL (KG/S)	700	1608.82	1608.12	0.04
	701	1710.95	1701.98	0.53
	702	1701.02	1694.88	0.36
	703	1715.82	1705.79	0.59
	704	1691.55	1691.69	0.01
	705	1610.81	1607.08	0.23
	706	1716.31	1705.65	0.63
	707	1694.08	1693.16	0.05
	708	1604.30	1606.81	0.16
	750	585.65	565.91	3.49
TEMPERATURE OUTER SURFACE CHANNELS (K)	7000	609.52	609.66	0.02
	7010	596.33	598.38	0.34
	7020	603.99	607.20	0.53
	7030	592.58	593.00	0.07
	7040	610.94	610.97	0.01
	7050	607.96	610.85	0.47
	7060	592.02	593.21	0.20
	7070	609.14	609.29	0.02
	7080	612.74	611.11	0.27
	7500	567.31	569.17	0.33
MASS FLOW DOWNCOMER & LOOPS (KG/S)	209	5245.58	5205.11	0.78
	309	5258.58	5205.11	1.03
	409	5249.15	5208.69	0.78
	DOWNCMR	15639.36	15581.11	0.37
PRESSURE PRESURIZER (Pa)	410	15470315	15452974	0.11
PUMP HEAD FOR THE PUMPS (Pa)	201	446876.7	451951.8	1.12
	301	441519.9	451951.8	2.31
	401	447093.0	452022.1	1.09

The mass flow rates through the channels of the core, show that RELAP5 and TRACE are in good agreement (see Table 1 combined with Figure 13).The relative differences in the channels for the fuel assemblies are below 0.63%; those in the core bypass channel is slightly higher, i.e. about 3.5%. This is due to the fact that the arrangement of the inlet of the core region is somewhat different in two input

models. In RELAP5, all the core channels are connected in the same way, to the outlet of the component describing the lower plenum. In TRACE the pipes for the fuel assemblies are connected to the side junctions of the component for the lower plenum region, whereas the bypass is linked to the outlet junction; so more flow is expected to go to the bypass.

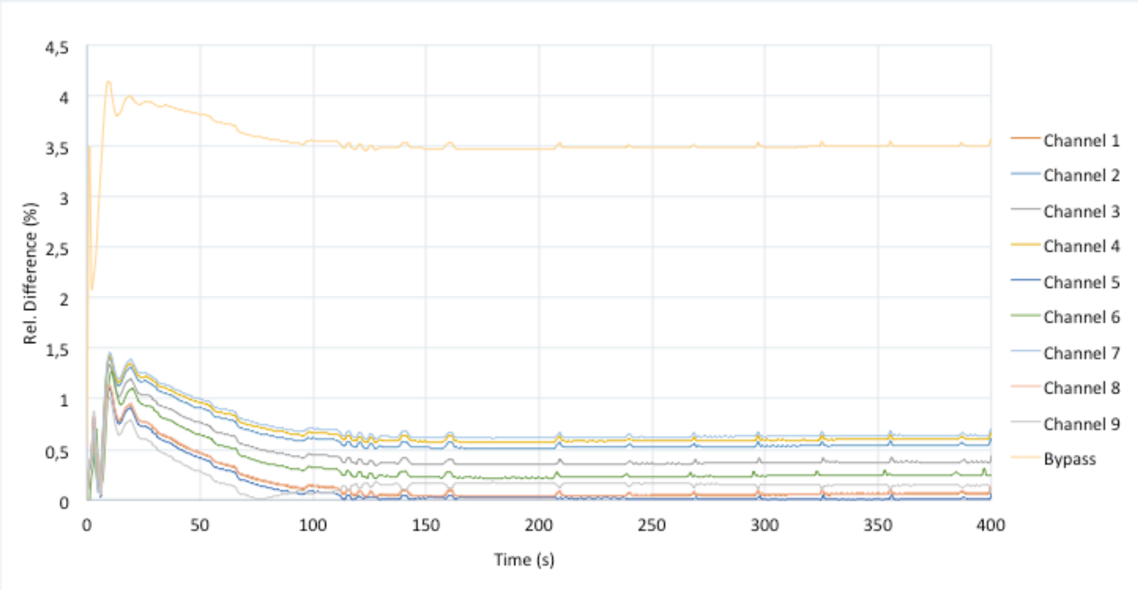

Figure 13 – Mass flow relative differences between RELAP5 and TRACE.

In Figure 14, the relative differences between the outer surface temperatures of the core channels are plotted. The maximum relative difference is 0.53%, as reported in Table 1.

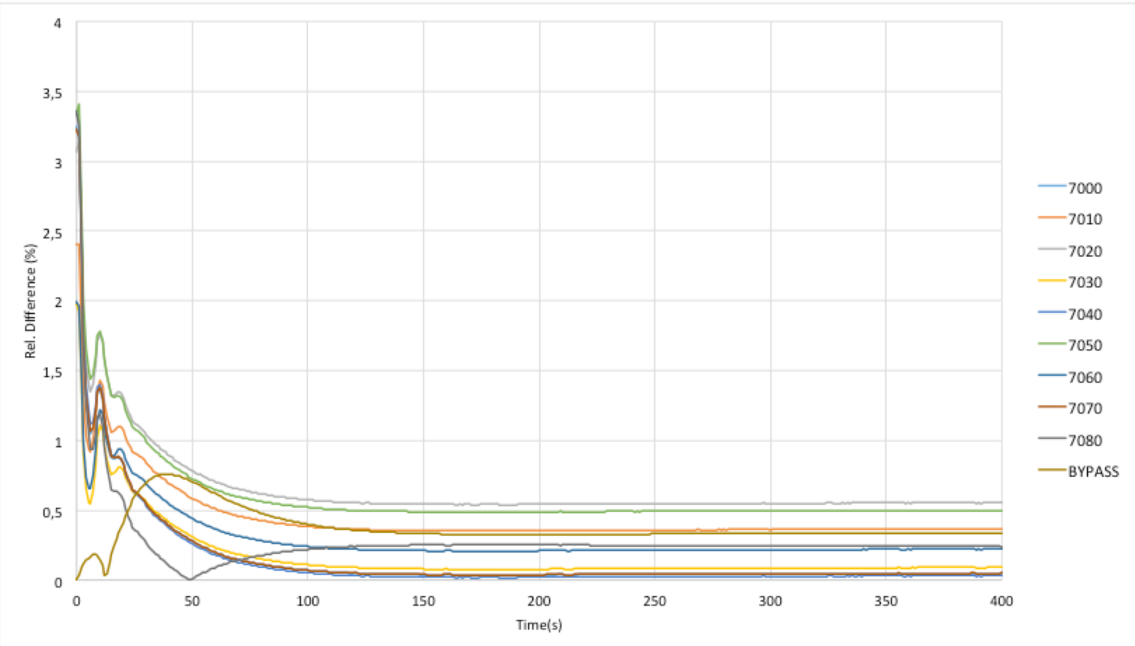

Figure 14 – Temperature relative difference between RELAP5 and TRACE in the outer part of the channels.

The mass flow rate in the loops and in the downcomer are compared. From Table 1 and Figure 15, the discrepancies for the three loops are relatively small: about 0.78% in two loops and about 1.03% in the third one. In the downcomer, the RELAP5 and TRACE models lead to the very close values of mass flow rate.

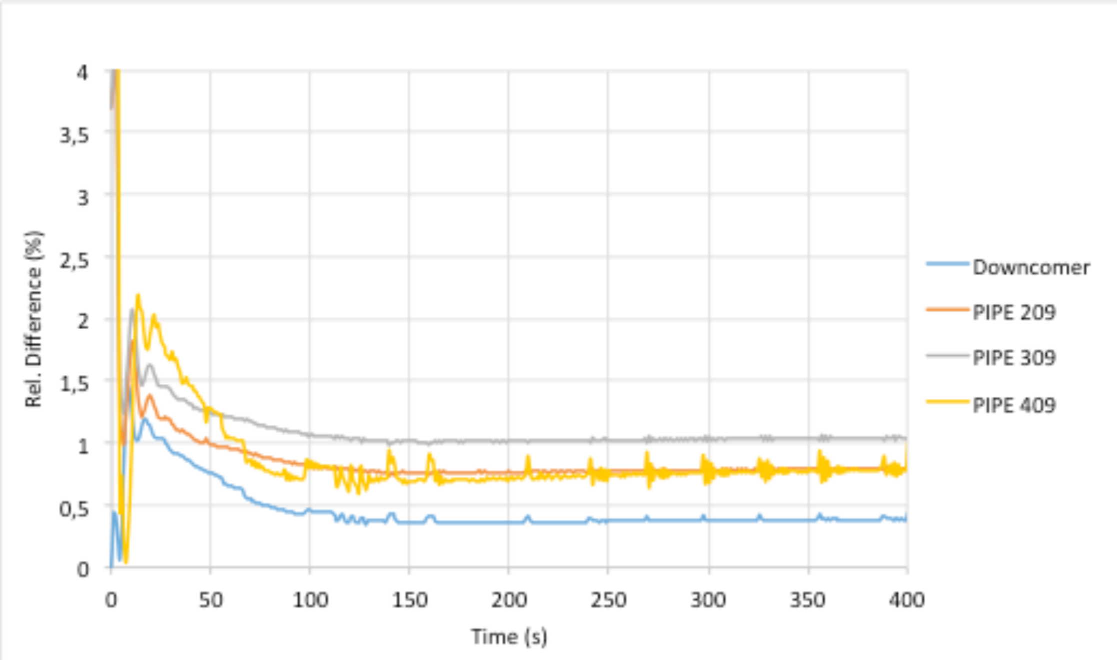

Figure 15 – Mass flow relative differences between RELAP5 and TRACE for the loops and the downcomer.

The difference in pressure is negligible, being about 0.11% (Table 1 and Figure 16).

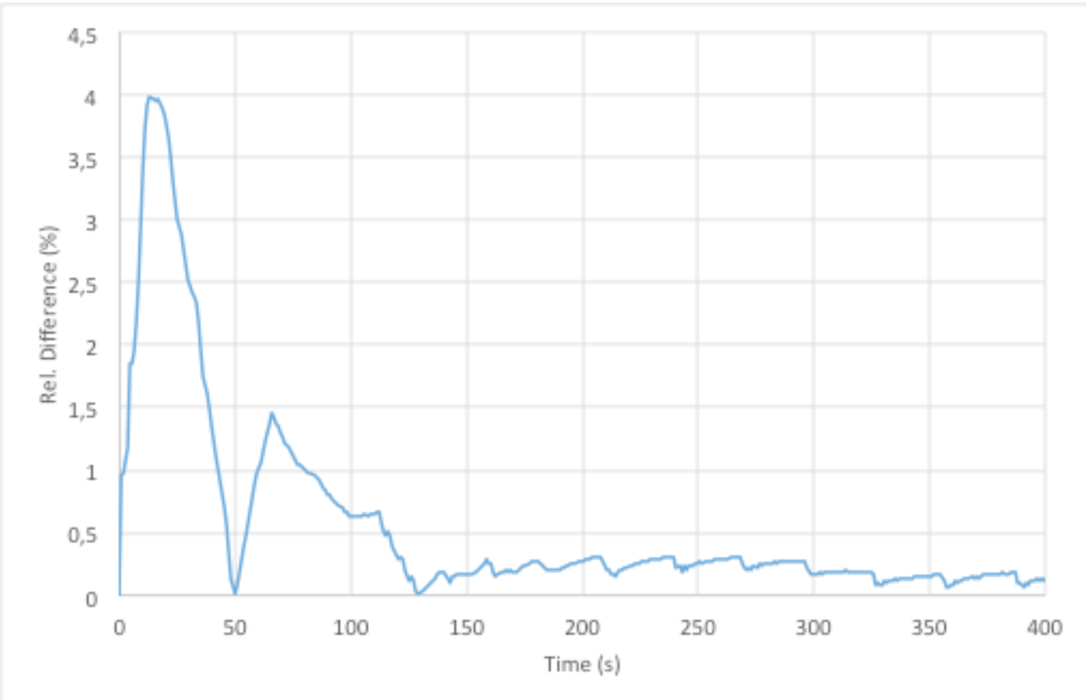

Figure 16 – Pressure relative difference between RELAP5 and TRACE in the pressurizer.

The relative differences in the pump head calculated with the RELAP5 and TRACE models, can be considered satisfactory (from Table 2, the maximum value is 2.31%). Comparing Figures 17 and 15, the analysis of the cooling loops and the pumps are consistent with each other. In fact the pump head related to two loops is similar, while a slightly higher difference is found in the third loop.

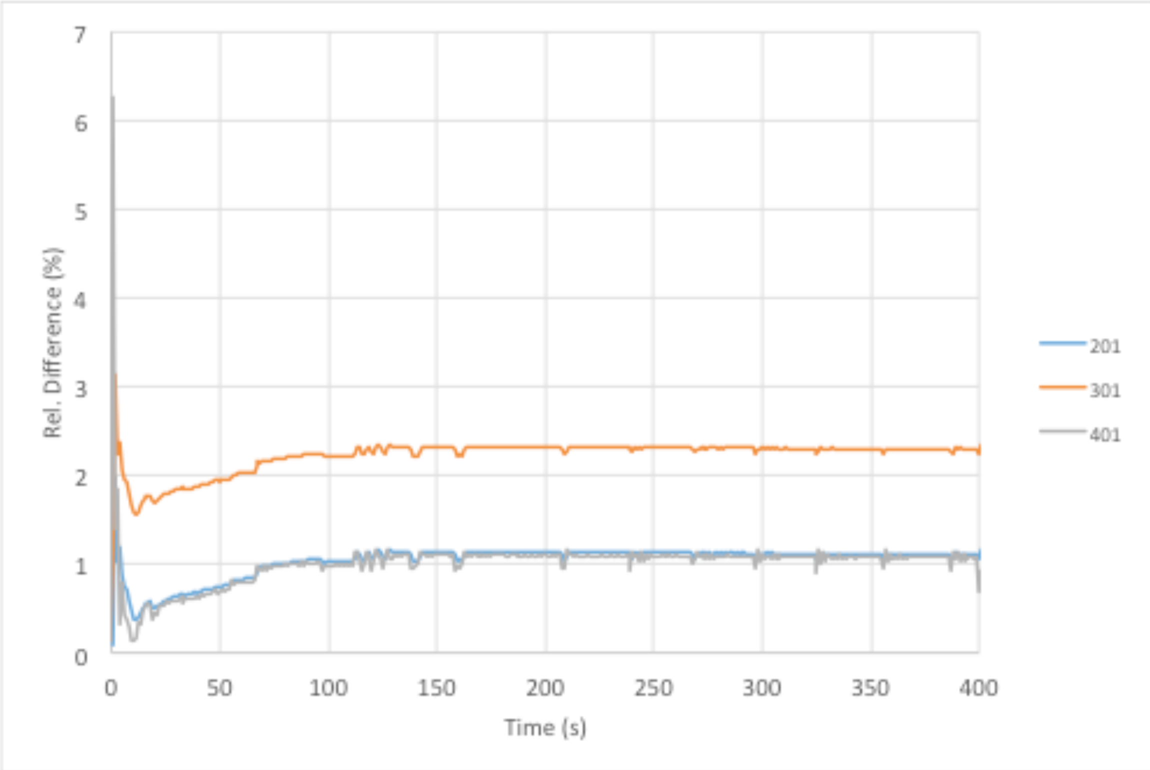

Figure 17 – Pump Head relative difference between RELAP5 and TRACE.

4.1.2 Comparison between the RELAP5 model and the TRACE model with the VESSEL component

The discrepancies for the converged parameters obtained with the RELAP5 input model and the TRACE input model using the VESSEL component, can be found in Table 2.
In this case, the mass flow rates through the channels in the core cannot be compared, since only one ring and one azimuthal sector were specified in the VESSEL component of the TRACE model.

Table 2 – Converged stand-alone, steady-state calculation: comparison between the RELAP5 input mode and the TRACE input model with the VESSEL component.

RELATIVE DIFFERENCES FOR DIFFERENT PARAMETERS				
PARAMETER	**HYDRAULIC ID**	**TRACE**	**RELAP5**	**DEVIATION (%)**
TEMPERATURE OUTER SURFACE CHANNELS (K)	7000	604.02	609.66	0.92
	7010	597.90	598.38	0.08
	7020	601.52	607.20	0.94
	7030	596.05	593.00	0.52
	7040	604.65	610.97	1.03
	7050	603.32	610.85	1.23
	7060	596.17	593.21	0.50
	7070	603.85	609.29	0.89
	7080	605.47	611.11	0.92
	7500	585.16	569.17	2.81
MASS FLOW DOWNCOMER & LOOPS (KG/S)	209	5213.18	5205.11	0.15
	309	5224.16	5205.11	0.37
	409	5216.12	5208.69	0.14
	DOWNCMR	15653.49	15581.11	0.46
PRESSURE PRESURIZER (Pa)	410	15443254	15452974	0.11
PUMP HEAD FOR THE PUMPS (Pa)	201	459293.5	451951.8	1.12
	301	454834.5	451951.8	2.31
	401	459675.0	452022.1	1.09

The difference in the temperatures of the outer surface of the heat structures used for the core region are displayed in Figure 18. The estimation of the relative differences are equal to: 2.81% for the heat structure assigned to the bypass channel; and between 0.08% and 1.23% for the heat structures used for the nuclear fuel (as reported in Table 2).

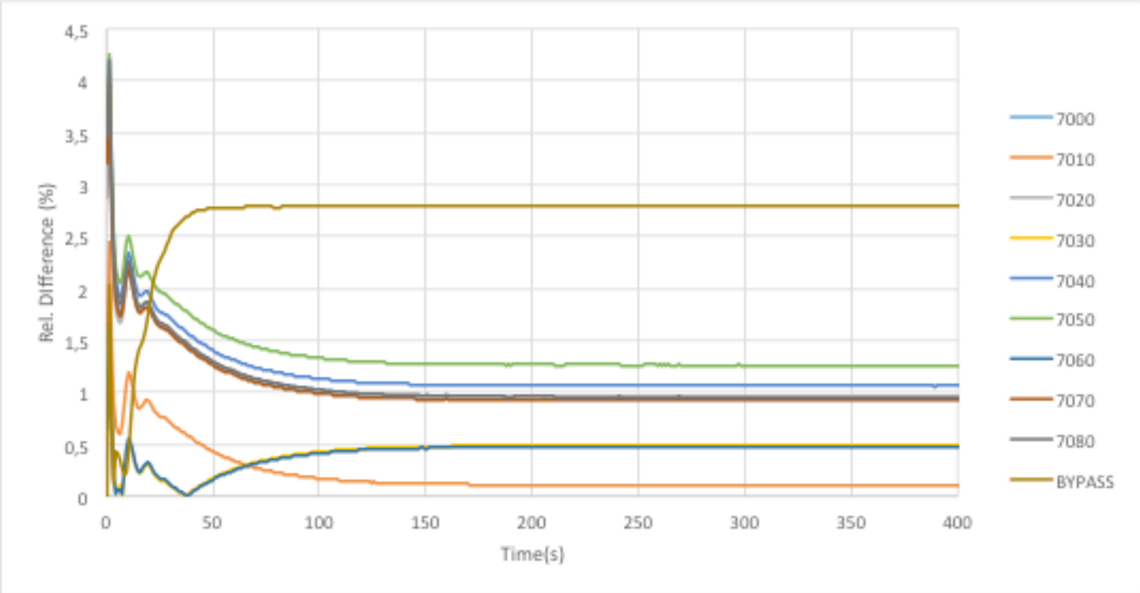

Figure 18 - Temperature relative differences between RELAP5 and TRACE for the channels modeling the fuel.

As regards the mass flow rates through the loops and the downcomer, the relative differences are small (Figure 20). The maximum value is about 0.46% and corresponds to the downcomer. Again, the results for two loops are very similar, while the difference in one of the loops is a little larger (0.37%, yet acceptable).

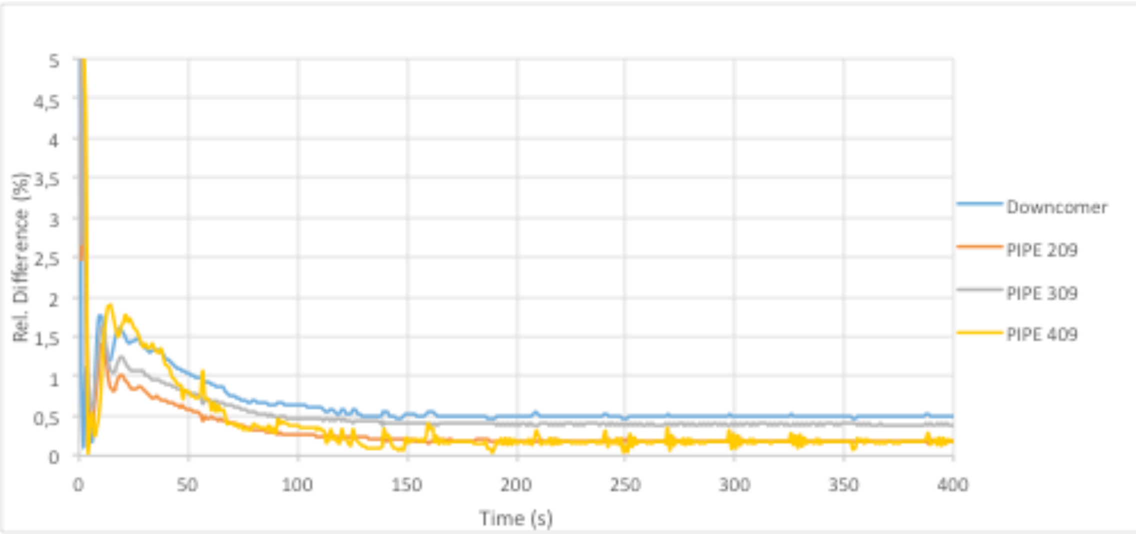

Figure 19 – Mass flow relative differences between RELAP5 and TRACE for the loops and the downcomer.

Small, negligible differences are also calculated for the pressure in the pressurizer, as shown in Figure 20.

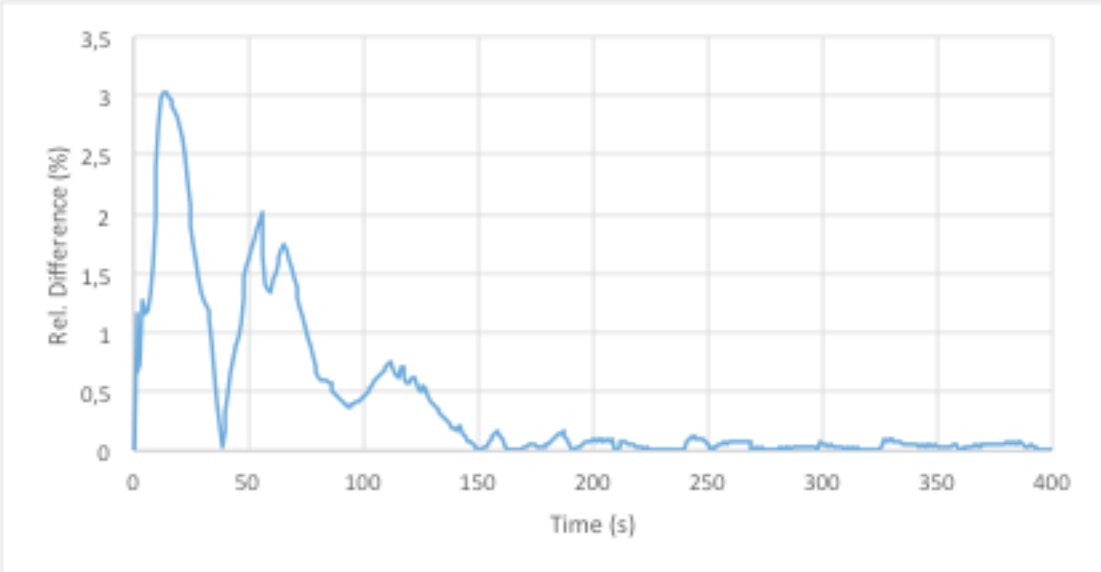

Figure 20 – Pressure relative difference between RELAP5 and TRACE in the pressurizer.

In Figure 21 the relative differences in pump head reach a maximum value of 1.69% for the loop with the pressurizer, which is slightly better than the case of the TRACE model without VESSEL component (see Figure 17).

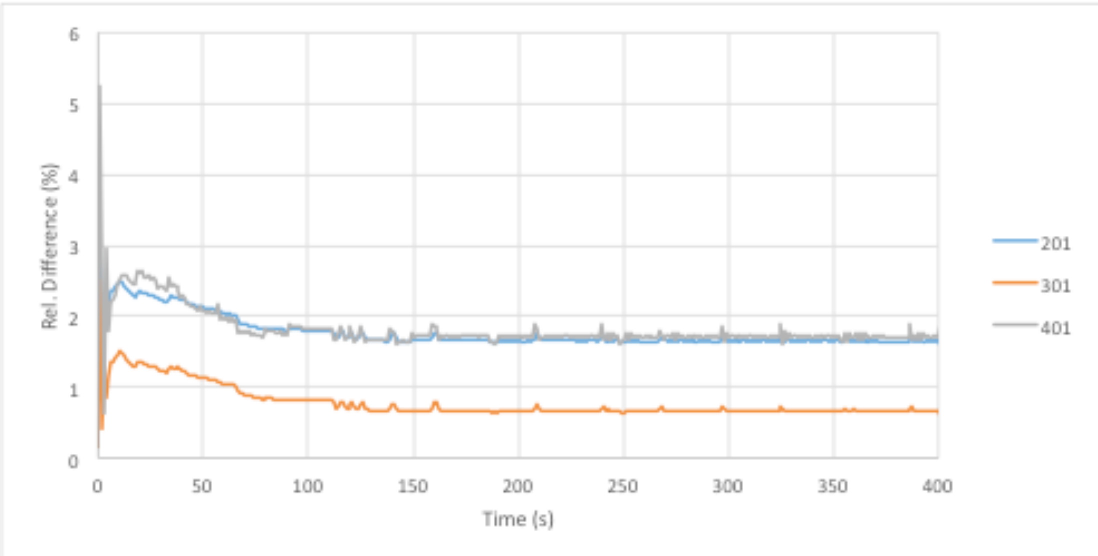

Figure 21 – Pump head relative difference between RELAP5 and TRACE for the pumps in the loops.

4.1.3 Verification of the Boron Injection System.

A test on the Boron Injection System is conducted by changing, after the steady-state has been achieved, the borated mass flow delivered to the primary system. The Boron injection with respect to time is reported in Table 3.

Table 3 – Boron injection for the stand-alone calculation with RELAP5 and TRACE.

BORON INJECTION	
Time (s)	**Mass Flow (kg/s)**
0	4.241
0.1	4.241
400	4.241
400.01	20
500	20
500.01	4.241
600	4.241

The test is performed under simplified, dummy conditions. In fact, no reactivity feedback from the variation of Boron is included, so that the injection will not have any impact on the reactor parameters, but the Boron content.

Figure 22 shows the Boron content in the reactor cooling system, at the connection with the Boron Injection System. Accordingly, Boron is added to the primary coolant at a constant rate and increases. At 400 seconds, more borated mass flow rate is delivered from the Boron Injection System, and the rise of boron in the primary coolant becomes faster. At 500 seconds, the boron injection is decreased, and the change of boron content slows down. The boron injection is then mimicked in a reasonable way, and the three models give the same results.

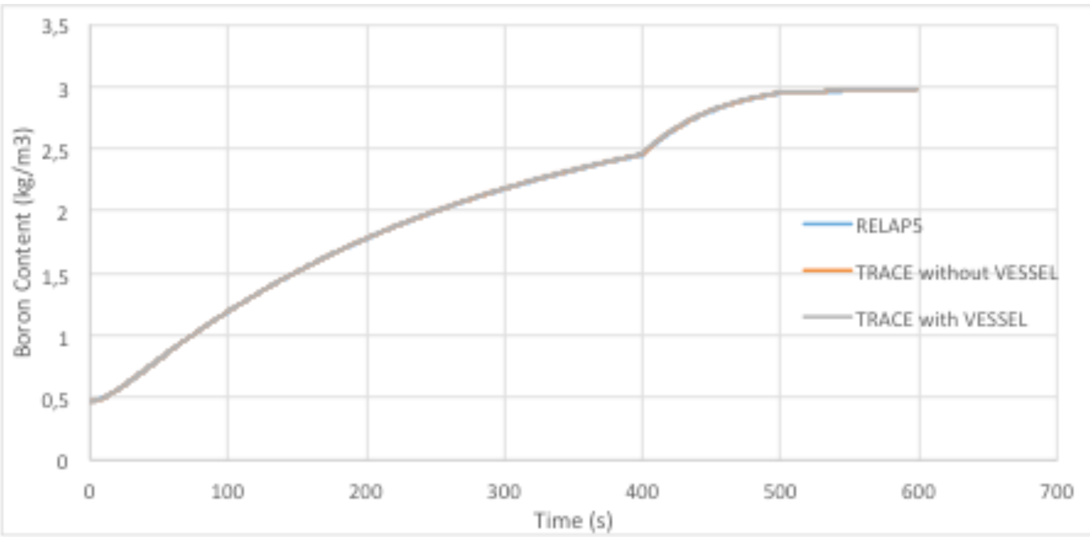

Figure 22 – Boron content on the first node after the Boron Injection System for the three models.

Figure 23 and Figure 24 show the Boron content at the inlet and outlet of the core respectively. In both places, the model with the VESSEL component has had a closer result to the one gotten in the original model RELAP5, especially for the values at the end of the simulation and having quite good approach until the injection of boron at time 400 seconds.

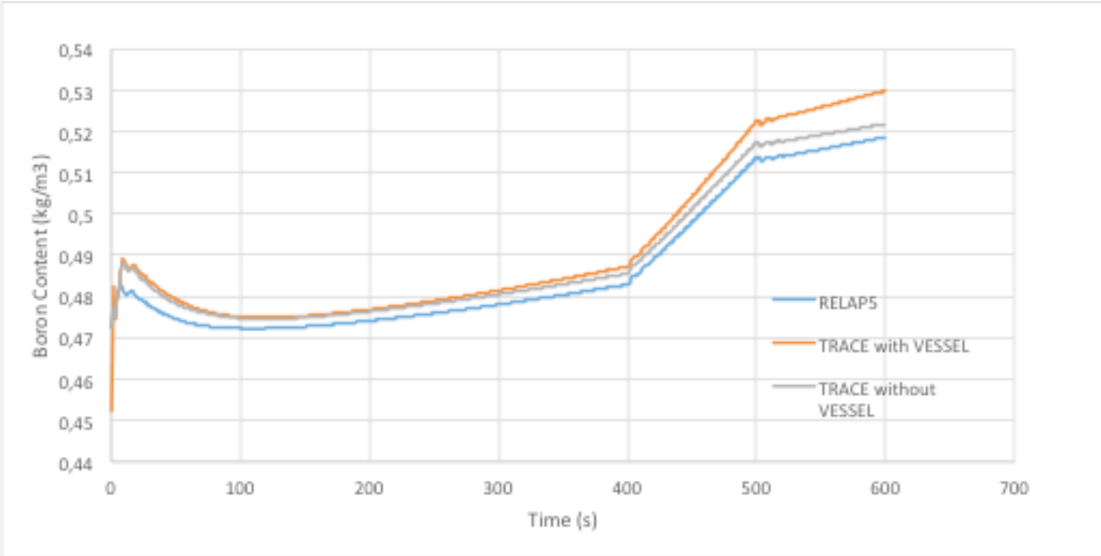

Figure 23 – Boron content at the inlet of the core for the three models.

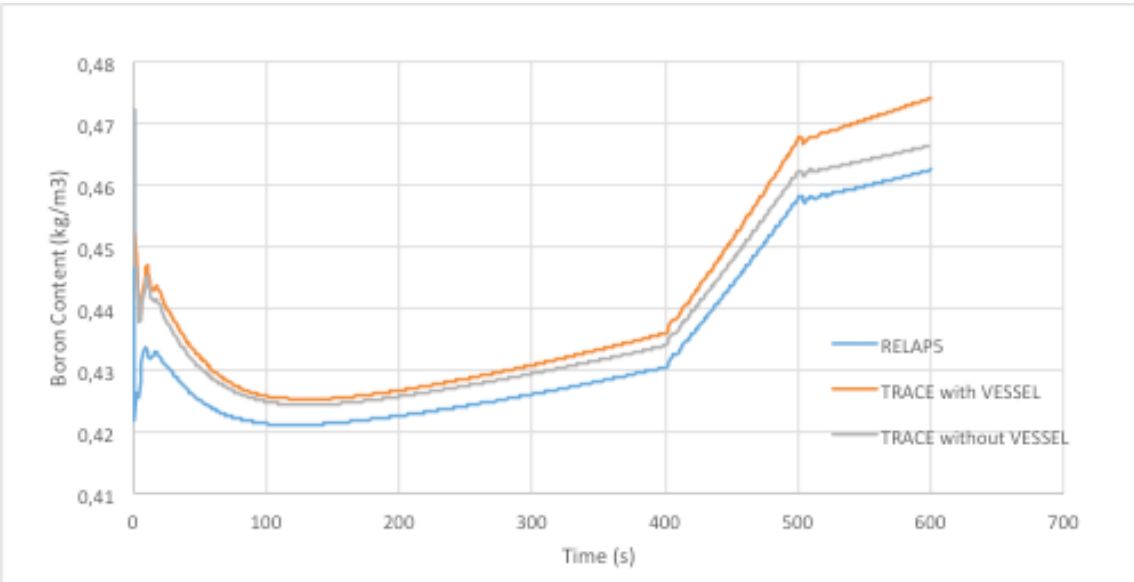

Figure 24 – Boron content at the outlet of the core for the three models.

In order to compare the results in a good manner, the relative differences for each of the plots above has been plotted. The lower relative difference has been gotten, as expected, for the node after the Boron Injection System. Then, the results for the model with the VESSEL are lower than 1% for both inlet and outlet. The results for the model without the VESSEL model are worse, with up to 2.51% of difference.

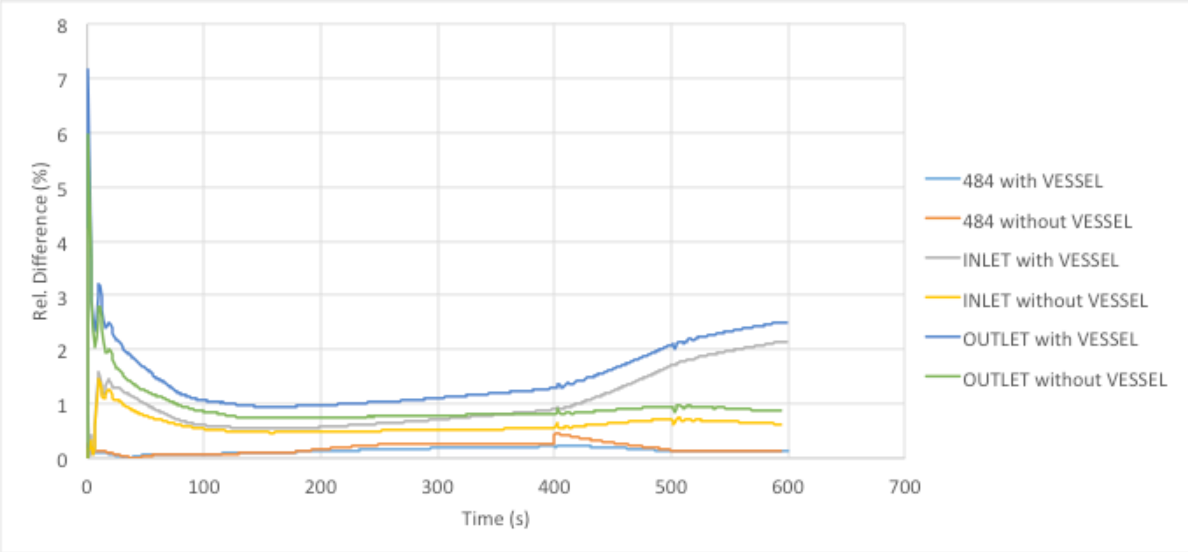

Figure 25 – Relative differences of the boron content.

4.2 Coupled Steady State (CSS)

During this simulation the both thermo-hydraulic codes are coupled with PARCS. At the end of the simulation the system has reached steady state conditions, according to the input information provided by PARCS and the thermo-hydraulic codes. In this section the axial and radial power shape will be compared.

4.2.1 Comparison between the RELAP5 model and the TRACE model without the VESSEL component.

Figure 26 shows the axial power shape evaluated with PARCS coupled to the RELAP5 input model and the TRACE input model without the VESSEL component, and Figure 28 shows the discrepancies. The results are quite similar for both cases. The first and last nodes have been removed since they are related to the bottom and top reflectors.

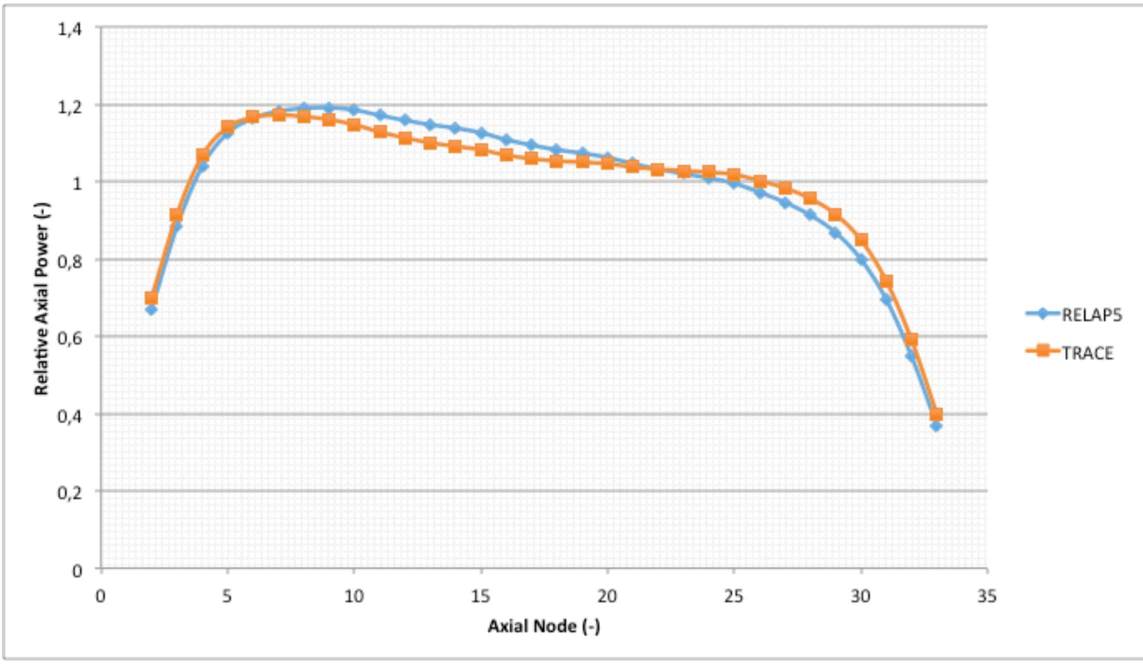

Figure 26 – Axial Power Shape for Coupled Steady State.

The differences in axial power between the two models, decreases in the lower part of the core. They increase from the 6th and to 14th axial node (up to a 4%) and decreases between the axial nodes 14 and 22. In the upper part of the core, above the 22th node, the difference grows again, up to 8% in the last node, as Figure 27 shows.

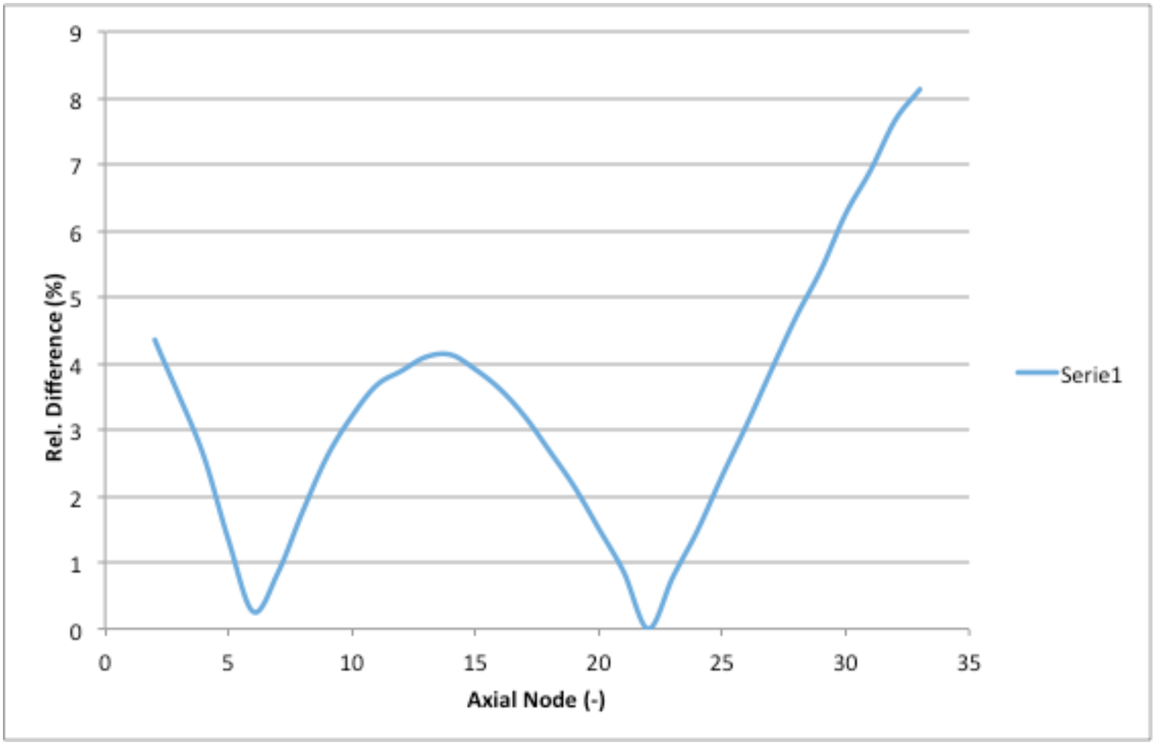

Figure 27 – Relative difference in the axial power shape for CSS for the model without the VESSEL component.

Figure 28 shows the differences between RELAP5 and TRACE in terms of radial power distribution. Even though, on average, good results are obtained, the radial power for some fuel assemblies can differ significantly. In particular, in the periphery of the core, on the left side, where the HTSTRs number 7010 and 7020 are placed (see Figure 7). This may also be affected by the discrepancies in the mass flow rate in the by-pass (see section 4.1.1).

					1.54%	1.75%	2.06%	0.88%	0.00%					
			1.94%	0.44%	0.64%	0.63%	0.64%	1.16%	2.31%	1.96%	0.90%			
		1.72%	0.16%	0.95%	1.11%	2.10%	2.74%	2.78%	2.59%	3.30%	2.80%	1.48%		
	2.53%	0.56%	1.70%	1.03%	0.13%	0.59%	2.11%	1.16%	1.33%	2.80%	4.57%	2.77%	0.78%	
	1.53%	0.05%	0.52%	0.77%	0.41%	1.00%	0.57%	1.33%	0.81%	2.40%	2.75%	3.23%	1.61%	
3.64%	1.21%	0.18%	0.94%	0.20%	0.44%	0.30%	0.77%	0.51%	1.15%	1.22%	1.05%	2.16%	1.71%	0.71%
4.57%	2.20%	0.20%	1.21%	0.23%	0.28%	0.40%	0.45%	0.52%	0.22%	0.77%	0.35%	1.89%	0.22%	1.94%
5.99%	4.09%	0.36%	0.86%	1.28%	0.28%	0.02%	0.82%	0.10%	0.05%	0.62%	0.15%	1.22%	2.11%	3.53%
5.73%	3.16%	1.06%	1.90%	0.77%	0.74%	0.06%	0.03%	0.12%	0.41%	0.35%	1.29%	0.11%	1.70%	3.66%
5.38%	3.05%	1.92%	2.22%	1.24%	0.39%	0.46%	0.04%	0.29%	0.09%	0.86%	1.87%	1.37%	2.08%	3.83%
	3.60%	1.84%	1.30%	0.63%	0.69%	0.07%	0.47%	0.21%	0.39%	0.25%	0.94%	1.54%	2.97%	
	3.99%	1.94%	0.17%	0.57%	0.87%	0.24%	0.71%	0.07%	0.56%	0.19%	0.22%	1.53%	3.46%	
		2.99%	1.76%	0.33%	0.64%	1.96%	2.61%	2.14%	0.95%	0.13%	1.35%	2.62%		
			2.73%	0.60%	1.18%	1.50%	0.38%	1.61%	1.39%	0.33%	2.43%			
					0.63%	0.60%	1.14%	0.55%	0.46%					

Figure 28 –Radial Power Shape relative difference for the Coupled Steady State Calculations for the model without the VESSEL component.

4.2.2 Comparison between the RELAP5 model and the TRACE model with the VESSEL component

The outcome from the comparison between the RELAP5 model and the TRACE model with the VESSEL component is such that significant differences can be found (Figure 29 and 30). The lack of good results may be due to the inherent geometry of the *VESSEL* and the use of only 1 channel for the VESSEL component in order to model the core. As in the case discussed in the previous subsection, the maximum relative difference belongs to the nodes in the upper part of the core (about 20%).

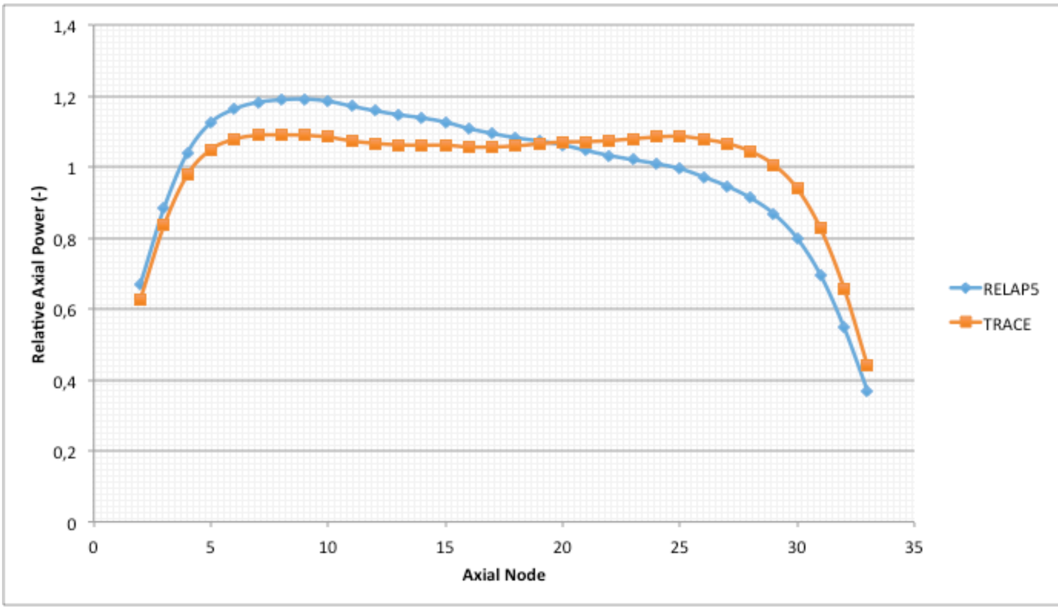

Figure 29 – Axial Power Shape for Coupled Steady State with the model using the VESSEL component.

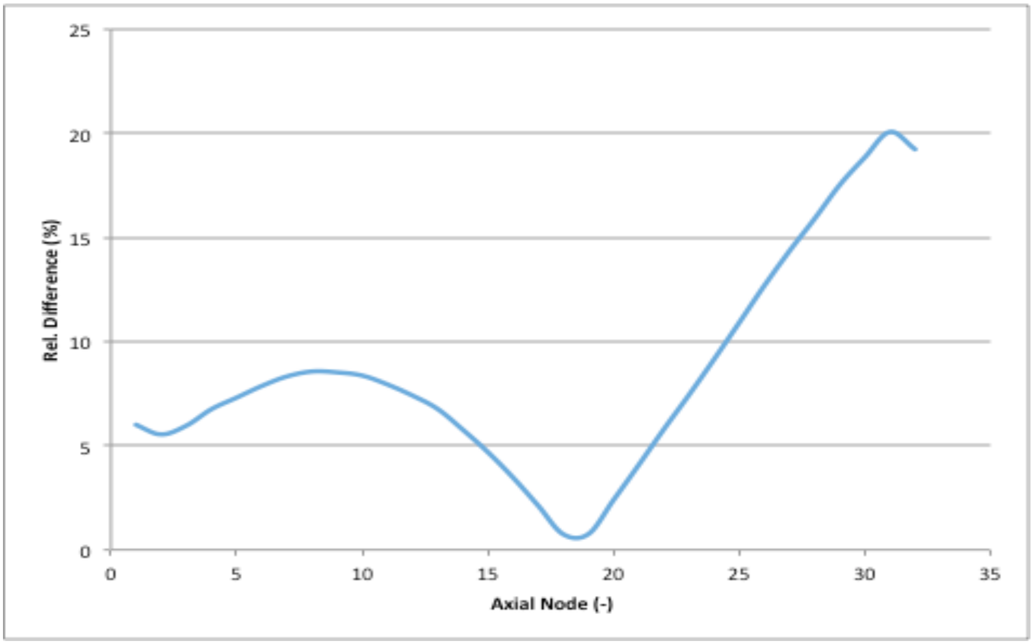

Figure 30 - Relative difference in the axial power shape for CSS for the model with the VESSEL component.

Figure 31 shows the differences in radial power. Such differences can be up to 10% in the periphery of the core.

					1.80%	0.65%	0.48%	2.08%	4.41%					
			2.20%	2.91%	3.24%	2.51%	0.96%	1.25%	6.31%	7.49%	7.78%			
		2.39%	3.23%	3.80%	2.21%	2.60%	2.19%	2.42%	3.48%	8.41%	9.43%	8.68%		
	1.23%	2.61%	4.50%	3.05%	0.42%	0.14%	0.69%	0.13%	0.88%	3.99%	9.71%	9.05%	7.07%	
	1.18%	2.50%	2.32%	0.95%	0.06%	0.02%	0.78%	0.07%	0.03%	1.64%	3.55%	7.63%	6.14%	
1.64%	0.32%	1.19%	0.99%	0.74%	0.62%	1.14%	0.83%	1.11%	0.31%	0.45%	0.20%	1.94%	4.20%	4.25%
3.88%	2.80%	0.76%	2.22%	1.55%	1.89%	1.43%	1.36%	1.50%	1.79%	1.32%	2.11%	0.12%	1.71%	1.33%
6.84%	5.65%	1.75%	2.49%	3.15%	2.22%	1.92%	0.91%	2.06%	2.29%	3.12%	2.45%	1.82%	5.43%	0.87%
7.05%	5.23%	2.94%	3.93%	2.83%	2.90%	2.32%	2.20%	2.31%	2.92%	2.98%	4.10%	3.17%	5.48%	2.21%
6.24%	5.07%	3.91%	4.16%	3.16%	2.29%	2.41%	2.01%	2.39%	2.30%	3.27%	4.50%	4.29%	5.06%	2.20%
	4.30%	2.98%	2.81%	2.23%	2.06%	1.30%	1.87%	1.25%	1.98%	2.23%	3.07%	3.56%	4.59%	
	1.99%	0.25%	0.85%	1.43%	1.50%	0.63%	0.43%	0.51%	1.29%	1.22%	0.86%	0.43%	2.14%	
		0.63%	1.97%	0.26%	1.76%	4.12%	5.24%	4.32%	2.10%	0.28%	1.73%	0.66%		
			0.06%	3.14%	6.61%	8.01%	7.24%	8.12%	6.83%	3.45%	0.29%			
					6.04%	6.53%	6.23%	6.58%	6.21%					

Figure 31 – Radial Power Shape relative difference for the Coupled Steady State Calculations for the model using the VESSEL component.

Statistics of both coupled steady state calculations is summarized in table 4, where the mean deviation of all the values for both the radial and axial shape power are included. The model without the VESSEL component is proven to have better performances.

Table 4 – Deviations of the axial and radial power distribution, between RELAP5/PARCS model and TRACE/PARCS with or without VESSEL component.

MEAN DEVIATION (%)			
DIRECT CONVERSION		USING VESSEL	
Radial	Axial	Radial	Axial
1.35 %	2.08 %	1.81 %	5.35 %

4.3 Coupled Transient Calculations

The transient scenario used for comparing the several coupled models, consists of a time-dependent boron injection in the primary coolant system according to Table 5. As discussed in section 1.1, Since Boron is a neutron absorber, so the variation of Boron content in the primary coolant can vary the power level of the reactor. In particular, a possible increase/decrease of the amount of Boron leads to a decrease/increase of the power.

Table 5 – Boron injection during the Coupled Transient.

BORON INJECTION	
Time (s)	**Mass Flow (kg/s)**
0	4.241
0.1	4.241
100	4.241
100.01	20
200	20
200.01	4.241
300	4.241

4.3.1 Calculation with the RELAP5/PARCS model

Figure 32 shows the results using the RELAP5/PARCS model. The initial average content of boron in the reactor is 631 ppm. From time t=0 to time t=100 seconds, water with a boron concentration of 4000 ppm is delivered at a constant rate, increasing the overall amount of boron in the system (orange line). In view of this, the power decreases. At time t=100 seconds, a higher borated mass flow rate is injected for a period of 100 seconds. So the change of boron in the reactor becomes faster and the power is reduced at a faster pace. At t=200 seconds, the borated mass flow rate is decreased to the initial value: as a consequence, the boron amount increases and the power level is reduced according to a slower slope.
The initial oscillations are related to the initialization of the calculation. The oscillations during the transient are due to the interplay between the variation of the injected boron and the resulting adjustment of the power level.

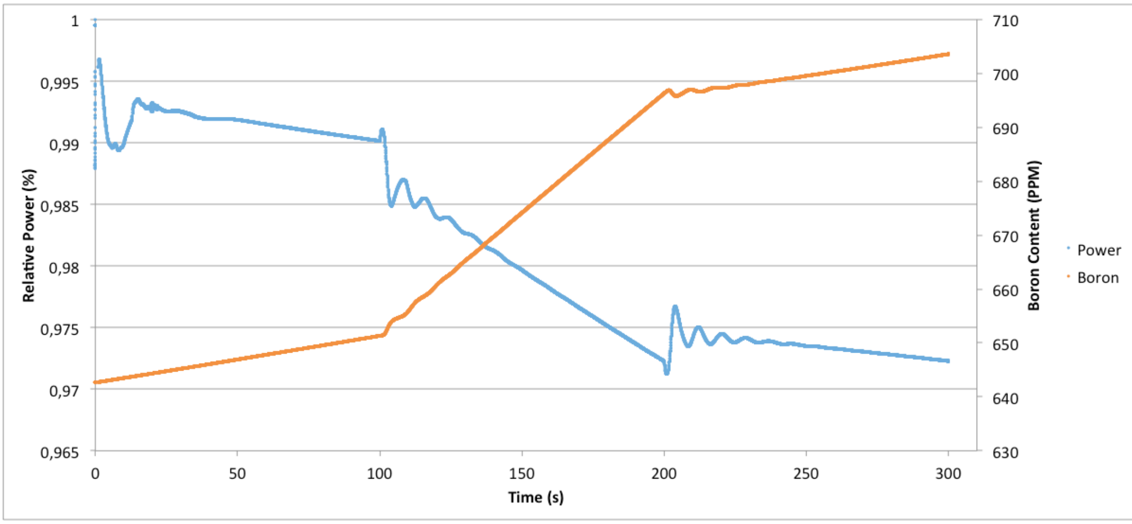

Figure 32 – Relative power (left axis) and boron content (right axis) during the transient in RELAP5/PARCS.

4.3.2 Coupled calculation with the TRACE model without the VESSEL component

The coupled transient calculated with the TRACE model without the VESSEL component, is displayed in Figure 33, and it is similar to the one calculated with RELAP5/PARCS. The differences between the two simulations along the transient, are given in Figure 34.
The differences between the rates at which Boron is injected into the system, are between 0.5 and 3.5%. During the first phase (between 0 and 100 seconds) the TRACE boron content phase of the transient is rather flat in comparison with the one given by the RELAP5 model. In the second part of the simulation, the increase of boron from TRACE is somewhat slower, and reaches a lower value at 200 seconds. In the last part of the transient between 200 and 300 seconds, both the simulations have a similar slope and the differences stabilizes at around 3.5%.As regards the power, the discrepancy between TRACE/PARCS and RELAP5/PARCS increases in the first part; then they becomes quite constant, around 1%.

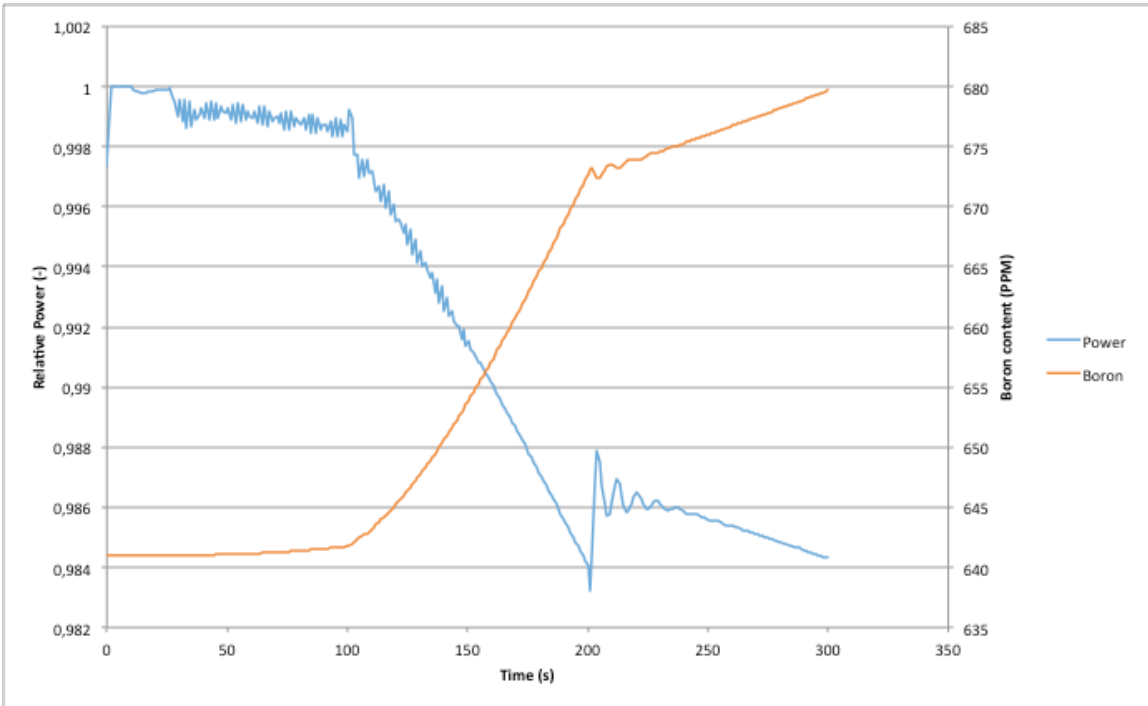

Figure 33 - Relative power (left axis) and boron content (right axis) during the transient in TRACE/PARCS.

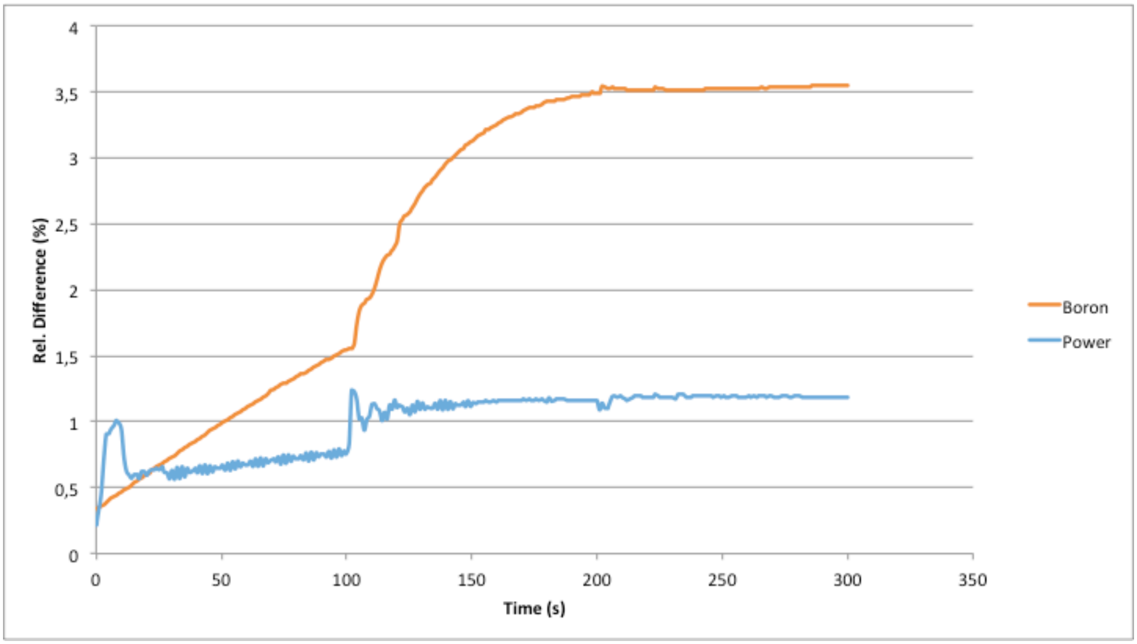

Figure 34 – Relative difference for power and boron compared to RELAP5.

4.3.3 Coupled calculation with the TRACE model with the VESSEL component

Although the share the same shape, the model with the VESSEL component has been run from a previous restart file that has actually converged in the CSS and therefore we can not see the fluctuations at the beginning of the transient. The boron content for the first 100 seconds still increases at a lower rate than the original RELAP5 model.

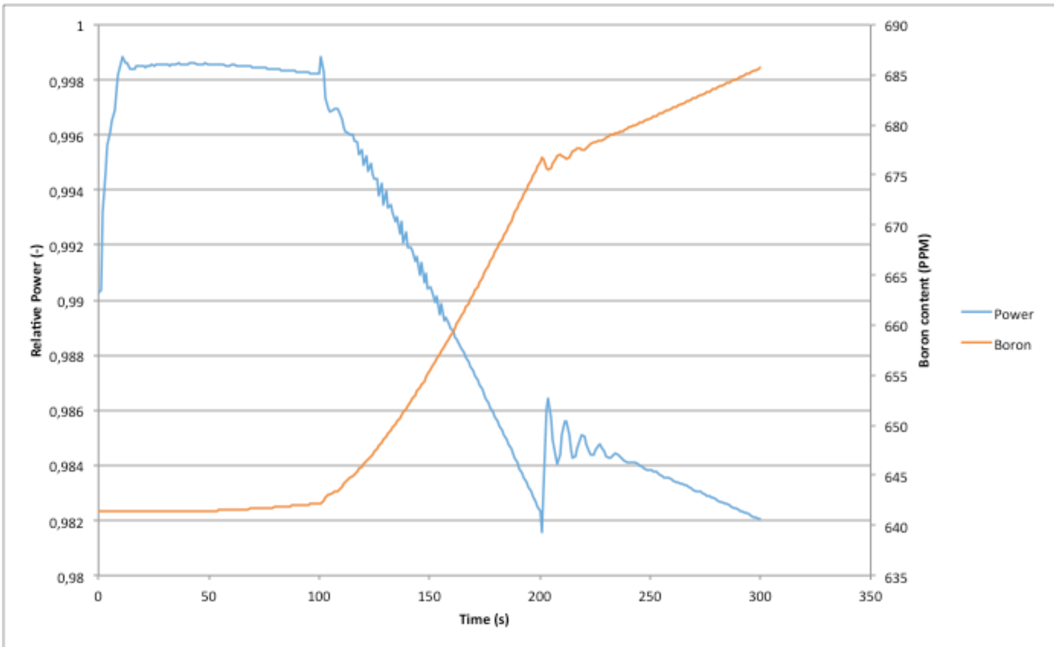

Figure 35 - Relative power (left axis) and boron content (right axis) during the transient in TRACE/PARCS.

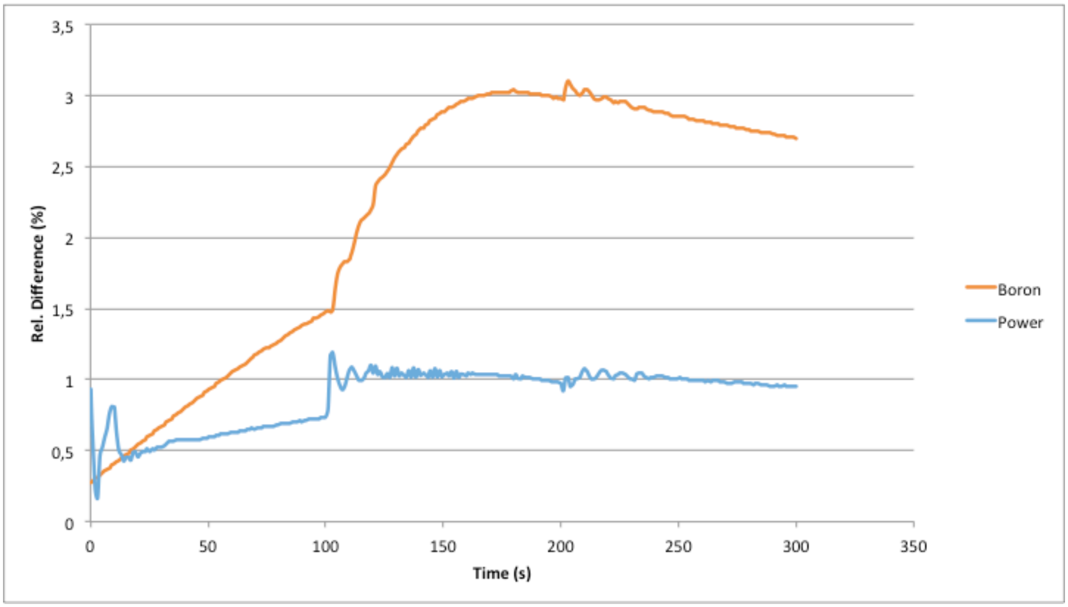

Figure 36 – Relative difference in boron and power compared to RELAP5.

Comparing Figure 36 and Figure 34, one can see the better capabilities of the model with the VESSEL component to properly simulate the transient just like RELAP5 does.

5 CONCLUSIONS

Two TRACE input models of a PWR have been built: the first one is directly obtained by converting an existing RELAP5 model; the second one is a modification of the first TRACE model, with the vessel region that is described with a special component available in TRACE, i.e. the VESSEL component.
In order to verify the correctness of the two new models, three types of calculations were performed and the results compared with the ones computed with the RELAP5 input deck. The three simulations are: a stand-alone, steady-state simulation; a steady-state simulation where the system codes are coupled to the neutronic code PARCS; a transient simulation coupled to PARCS, where the boron content in the primary coolant flow changes with time.
From the analyses, the two TRACE input models provide reasonable results with respect to the ones calculated with RELAP5.
For the first TRACE input model, the stand-alone, steady-state simulation gives mass flow rates in the core and in the primary loops that are very similar to the RELAP5 results. The relative differences are below 1%, but a higher value (3.40%) is estimated in the core by-pass channel; this suggests that improvements may be needed for that region. The temperature of the heat structures associated to the core is also well-reproduced: in fact the maximum relative discrepancy with RELAP5 is about 0.53%. Also, the behavior of the pumps, pressurizer and boron injection system are predicted close. Slightly higher differences were obtained for the pump heads, i.e. between 1% and 3%.
From the coupled, steady-state simulation, the first TRACE input model and the RELAP5 input model lead to comparable results in terms of axial and radial power distributions. In fact the mean deviation for the axial power profile is 2.08%, while it is 1.35% for the radial power.
In the coupled, transient simulation, the results for the first TRACE input deck and the RELAP5 one are such that differences are below 3.52% for the boron content and 1.22% for the power.
As regards the second TRACE input model, where the VESSEL component is used, the results showed that refinements may be necessary. In fact such a model includes only one hydraulic channel for the core region. In the stand-alone, steady-state calculation, the total core mass flow rate is close to the RELAP5 one (the relative deviation is about 0.5%), but the relative differences in the core temperatures can reach almost 3%. Moreover, when considering the coupled, steady-state calculation, the core power distributions can be quite different from the RELAP5 one: the mean relative deviation of the axial power is about 5.35%, while the mean relative deviation of the radial power is 1.81%. For the coupled, transient simulation, the comparison between the second TRACE input model and the RELAP5 one, indicates that a better approach than the first TRACE model has been gotten with 3.05% of mean relative deviation for the boron content and 1.16% for the power.

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