TRACE CODE VALIDATION FOR NATURAL CIRCULATION DURING SMALL BREAK LOCA IN EPR-TYPE REACTOR

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Stockholm, Sweden 2011
Abstract

The PWR PACTEL test facility was built in Lappeenranta (Finland) to gain experience in thermal-hydraulics behavior of vertical steam generators used by EPR (European Pressurized Water Reactor) during SBLOCA (Small Break Loss of Coolant Accident) transient, which involves natural circulation phenomenon. The benchmark, which consisted of blind and open part, offered a unique opportunity for code users to improve and test their knowledge and skills in developing the input deck models and performing calculations. For a purpose of this investigation, Royal Institute of Technology (KTH) has developed two TRACE code models.

The main point of this thesis is to study TRACE code performance during SBLOCA transient and sensitivity of the developed TRACE models for the time and space convergence, which is very important for transients involving natural circulation phenomenon. Four different nodalizations coarse, inter-medium, fine and fine-sliced (space convergence), are designed for both designed models, which are calculated with different maximum time steps (time convergence). The results assessment was made by comparisons of the main parameters e.g.: Pressure of upper plenum, Inlet/outlet temperature of Core/SGs, Collapsed water level in the core, among others. In addition, discussion about vertical SGs performance during natural circulation phenomenon and conclusions for both, code users and developers, are provided.
Acknowledgements:

I would like to express my gratitude and sincere appreciation to my advisor Dr. Tomasz Kozlowski for giving me the chance to carry out this project at Royal Institute of Technology, Stockholm, and for his guidance and encouragement through the course of this work.

I am especially indebted to my co-advisor Joanna Peltonen for her continuous support during models development. Many detailed discussions with her have invaluably shaped the evolution of this thesis.

Many thanks to Heikki Purhonen, Vesa Riikonen and their colleagues from Lappeenranta University of Technology for the benchmark experiment in PWR PACTEL test facility and making this project possible.

Aquest projecte està dedicat als meus pares Joan i Neus i a la meva germana Anna per tot l’amor que m’han demostrat sempre i que mai no podré agrair prou.

Stockholm, September 2011

Joan Bertran i Morancho
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Abbreviations

**BWR**: Boiling Water Reactor  
**BC**: Boundary Conditions  
**BOT**: Begin Of transient  
**CB**: Control Block  
**CL**: Cold Leg  
**ECCS**: Emergency Core Cooling System  
**EPR**: European/Evolutionary Pressurized Reactor  
**FDW**: Feedwater  
**HL**: Hot Leg  
**HPSI**: High Pressure Safety Injection  
**KTH**: Kungliga Tekniska Högskolan (Royal Institute of Technology)  
**LOCA**: Loss of Coolant Accident  
**NC**: Natural Circulation  
**NPP**: Nuclear Power Plant(s)  
**PACTEL**: Parallel Channel Test Loop  
**PWR**: Pressurized Water Reactor  
**PZR**: Pressurizer  
**RCP**: Reactor Coolant Pump  
**RPV**: Reactor Pressure Vessel  
**SBLOCA**: Small-Break Loss of Coolant Accident  
**SG**: Steam Generator  
**SL**: Steam Line  
**SS**: Steady State  
**SNAP**: Symbolic Nuclear Analysis Package  
**TH**: Thermal-Hydraulics  
**TRACE**: TRAC/RELAP Advanced Computational Engine  
**U.S.NRC**: United States Nuclear Regulatory Commission
Chapter 1

Introduction

After March 2011 Fukushima accident in Japan, many questions and doubts related to nuclear safety have raised worldwide, bringing about the cancellation and revision of nuclear programs in several countries. Nevertheless, many lessons are and will be learned from this disaster; therefore researchers, scientists and engineers must keep on reviewing safety protocols, designing safer reactors and carrying out safety analysis and experiments. Fukushima has revealed many deficiencies in currently operating Nuclear Power Plants (NPP), which must be solved and fixed in order to avoid in the future similar problems.

This thesis is another step on the path to a free of carbon emissions future, in which safe and clean nuclear energy should play an important role. Its main purpose is to validate a relatively new computational code –TRACE, which is said to be the code of the future - with a Small-Break Loss of Coolant Accident (SBLOCA) experiment under Natural Circulation conditions which was hold in 2009 in a test facility scaled from Generation III+ European Pressurized Reactor (EPR). Codes are nowadays the main tool of Nuclear Safety experts and it is fundamental to validate them in order to rely on their results.

In this introductory chapter a brief review of physical phenomena involved during a SBLOCA is presented. After that, the focus is put on PWR PACTEL test facility and the benchmark experiment, and finally the main objectives and motivation for this thesis are displayed.
1.1 Small-Break LOCA accident

Small-Break LOCA research started worldwide after the Three Mile Island accident in 1979, it resulted in construction of many test facilities, which analyze the TH phenomena associated with LOCA. Many reports were dedicated for detail investigation of physical phenomena involved with SBLOCA [1] [2]. Results show the dependence of many factors such as reactor design, break location, ECCS location, break size, boron concentration or non-condensable gases concentration. They also revealed the importance of reactor operator actions (human factor).

During SBLOCA the depressurization of primary system is less violent than during Large-Break LOCA. During the first seconds of depressurization, water almost immediately reaches the saturation conditions, changing into two-phase mixture and separation of liquid and vapor occurs (heat-transfer is imposed by multi-phase flow). Water level decrease in the vessel is a slow process; thus, core dry-out occurs only if Emergency Core Cooling Systems (ECCSs) are not available or operator makes a mistake. Typically SCRAM and High Pressure Safety Injection (HPSI) are initiated between 20 – 60 seconds because of high pressure in containment, low water level in the Pressurizer (PRZ) or low pressure in the Reactor Pressure Vessel (RPV). If the break is large enough, the pressure decrease continues because the HPSI isn’t efficient enough to cease inventory decrease until it reaches a value of saturation temperature, which is close to SGs water temperature. The pressure in the vessel is almost constant until water inventory in the vessel reaches the hot leg level; this results in second depressurization due to escaping to the break superheated steam.

At this point, procedures order to stop all the Reactor Coolant Pumps (RCP) to reduce loss of coolant rate, which causes that Natural Circulation (NC) is the main way of decay heat removal. NC is a passive cooling mechanism, which plays an important role in nuclear accidents with non-availability of RCP. The density difference between fluid flowing out from the core and water in the heat sink (U-tubes of Steam Generator) at a higher elevation induces passive and natural water flow. The efficiency of this cooling system depends on core power, heat sink temperature, elevation difference between hot and cold parts, and coolant inventory among others. During the NC transient core heat is removed by an efficient saturated two-phase mixture. The steam condensates in the SG’s U-tubes and returns as a water to the core (reflux mode). The reflux mode of the SG is very efficient and long lasting way of cooling the core.

If pressure in the reactor coolant systems is low enough, the subcooled water injected by HPSI pumps can finally exceed the rate of flow leaking out through the break and starting the plant recovery. Water levels in the upper plenum will increase again and rate of steam generation in the core will decrease; single-phase natural circulation will be established and will keep cooling the reactor unless non-condensable gases or loss of heat removal capability in SGs interrupts it (as it happened in Three Mile Island NPP accident).
1.2 PWR PACTEL Test facility

PWR PACTEL test facility [3] was first constructed in 1990 in Lappeenranta, Finland, to study Soviet-designed VVER-440 PWR operating since 1980s in the country. The facility was recently partially redesigned to gain experience in Thermal Hydraulics evolution, safety studies and accident management procedures for the EPR Nuclear Power Plant (NPP) Olkiluoto-3, at the moment under construction [4]. The Core, the Pressurizer (PRZ) and the Emergency Core Cooling Systems (ECCS) were slightly upgraded but were basically the same components, while loops, which originally had horizontal SGs, were completely remodeled and rebuilt. This research laboratory includes multiple transducers and instrumentation to obtain temperatures, pressures, differential pressures and flows in many locations of the primary loop in order to obtain extra data which would never be acquired in a NPP. A special focus is set on U-tubes to provide information about new SG construction and operation, as their performance during accidents requires further investigations. A picture representing PWR PACTEL test facility and main features are next presented in Table 1 and Figure 1:

<table>
<thead>
<tr>
<th>Reference power plant</th>
<th>PWR (EPR)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Volumetric scale</td>
<td>1: 405 Pressure Vessel</td>
</tr>
<tr>
<td></td>
<td>1:400 Steam Generators</td>
</tr>
<tr>
<td></td>
<td>1:562 Pressurizer</td>
</tr>
<tr>
<td>Height scale</td>
<td>1:1 Pressure Vessel</td>
</tr>
<tr>
<td></td>
<td>1:4 Steam Generators</td>
</tr>
<tr>
<td></td>
<td>1:1.6 Pressurizer</td>
</tr>
<tr>
<td>Maximum heating power</td>
<td>1MW</td>
</tr>
<tr>
<td>Maximum primary / secondary Pressure</td>
<td>8.0MPa / 4.65MPa</td>
</tr>
<tr>
<td>Maximum primary / secondary Temperature</td>
<td>300ºC / 260ºC</td>
</tr>
<tr>
<td>Maximum cladding temperature</td>
<td>800ºC</td>
</tr>
<tr>
<td>Number of primary loops</td>
<td>2</td>
</tr>
<tr>
<td>Steam Generator tube diameter</td>
<td>Ø19.05 x 1.24 mm</td>
</tr>
<tr>
<td>Average steam generator tube length</td>
<td>6.5m</td>
</tr>
<tr>
<td>Number of U-tubes in one steam generator</td>
<td>51</td>
</tr>
<tr>
<td>Number of instrumented U-tubes SG I / SG II</td>
<td>8 / 14 (51*)</td>
</tr>
<tr>
<td>Main material of components</td>
<td>Stainless Steel (AISI 304)</td>
</tr>
<tr>
<td>Insulation material</td>
<td>Mineral wool (aluminum cover)</td>
</tr>
</tbody>
</table>

Table 1: PWR PACTEL test facility design parameters [3]
Special attention needs to be focused on SG construction in Lappeenranta University of Technology in order to model EPR type SG, as they contain the main new changes compared to VVER’s SG. The main purpose of this study is that in the new vertical steam generators, the angle between hot leg outlet and U-tube inlet changes from 0 to 90 degrees, considerably affecting elevation and TH behavior, especially during the reflux mode.
1.2.1 Steam Generators description

PWR PACTEL Steam Generators (Figure 2) were built to simulate the behavior of a single EPR type SG. However, as described in the previous table, the height scaling factor for these components was about 1 / 4 compared to the real ones instead of the desired 1 / 1 due to a limited elevation of laboratory’s ceiling. As a result of that, the volumetric scaling approach wasn’t completely achieved, so the results could be slightly affected.

As said in the main description, each SG contains 51 heat exchange U-tubes (Ø19.05x1.24) of an average length of 6.5 meters in an equilateral triangular grid of 27.4mm of side length, while heat transfer area between primary-side volume of each SG and tube bundles is scaled 1/400. In the secondary side there is a flat metal slab dividing hot and cold part as in EPR steam generators, but the volume in the secondary side doubles the volumetrically scaled of boilers in EPR. Steam separators are not modeled as their influence in TH behavior should be insignificant. Each SG consists of independent feedwater (FDW) systems, and both steam lines are connected to the atmosphere where steam flows, providing the boundary conditions of secondary system. Due to imperfect manufacturing, pressure balancing between downcomer and steam volume is possible; minor cross flows between hot and cold parts could slightly affect the results.

Figure 2: PWR PACTEL steam generator general view [3]
1.2.2 Benchmark SBL-50 experiment

An experiment to study PWR PACTEL’s integral test facility behavior during SBLOCA transient under Natural Circulation (NC) conditions was carried out. It’s really significant to analyze the capability of passive core cooling with the revolutionary EPR type Steam Generators under Small Break Loss of Coolant Accident (SBLOCA) transients. NC mode and its cooling efficiency are affected by the total primary mass inventory availability, being able to bring about dangerous peak cladding temperature in the core. Thus, an experiment with the purpose of comparing vertical and horizontal SG performance in NC conditions with decreasing water inventory was designed [5].

The transient initially begins with Steady State (SS) operation, which was established for 1000 seconds at full inventory with coolant circulating through both loops. PRZ heaters were switched off and it was isolated by closing a valve placed on the pressurizer line trying to avoid changes in mass flow rate of primary side by fluid entering and escaping from the pressurizer. An orifice plate of 1mm of diameter placed between the loop seal and the cold leg connection to the vessel was used to simulate the break, representing 0.04% of Cold Leg (CL) cross-sectional area. This location causes maximum water leakage from the facility (and in a NPP could also affect the capability of coolant supplied by ECCSs to reach the core), so it is the most conservative. A tank was connected to the break to determine the break flow rate by measuring the inventory variation by condensing the leaking steam. Steady State initial conditions are presented in Table 2.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>VALUE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary side pressure</td>
<td>75 bar ± 1 bar</td>
</tr>
<tr>
<td>Secondary side pressures</td>
<td>42.0 bar ± 0.6 bar</td>
</tr>
<tr>
<td>Core Power</td>
<td>155 kW ± 6 kW</td>
</tr>
<tr>
<td>Pressurized Collapsed level</td>
<td>5.7 m ± 0.2 m</td>
</tr>
<tr>
<td>Steam Generators collapsed levels</td>
<td>3.9 m ± 0.12 m</td>
</tr>
<tr>
<td>SGI feedwater temperature</td>
<td>23 °C ± 1 °C</td>
</tr>
<tr>
<td>SGII feedwater temperature</td>
<td>19 °C ± 1 °C</td>
</tr>
<tr>
<td>Steam Generator feedwater flow rate</td>
<td>1.5 L/min ± 0.4 L/min</td>
</tr>
<tr>
<td>Loop mass flow rates</td>
<td>0.6 kg/s ± 0.14 kg/s</td>
</tr>
</tbody>
</table>

Table 2: Initial SS conditions for PWR PACTEL SBL-50 experiment [4]

The break valve was opened and transient started and measurements of main parameters were recorded. Secondary side water level in both SG was maintained to keep the pressure constant by adding FDW, but no other actions were taken during the whole experiment. However, an unexpected incident occurred during the experiment, and the break closed at about 7700 seconds due to an error in the valve control. It was noticed and fixed after 8960 seconds, so a quasi SS was set during that part of the transient. When the top part of the core commenced drying out and temperature in core outlet reached the value of 350°C, the experiment was terminated to avoid damage the test facility.
Results of PWR PACTEL SBL-50 experiment are briefly presented: the pressure fell until saturation temperature reached core outlet temperature, with a single-phase liquid constant mass flow rate circulating through the legs. As the break is considerably small and low energy loss rate escaped through it, the primary pressure kept far over secondary pressure during these 3000 seconds of transient. Secondary side pressure was kept constant during the whole transient (Figure 3).

![Figure 3: Pressure in upper plenum and steam generators VS time [4]](image)

When the water level in the upper plenum reached hot leg elevation, a two-phase mixture started to flow through hot leg and steam condensed in the SGs and a relatively steady type of flow was established 3000 seconds after begin of transient. Highest mass flow rate values were reached when 70% of total inventory mass was still in primary side (around 4000 seconds after transient started). Nevertheless, mass flow rate starts decreasing as inventory keeps escaping through the break (see Figure 4).
However, when mass inventory reached 50% of total initial primary mass, reflux condensation or boiler condenser mode started, and steam generated in the core was condensed in both SGs. Collapsed water level in the RPV was lower than hot leg connection, so steam was only circulating through HLs. If steam flow velocity was high enough, the condensate was dragged to the cold leg, but for low steam flow speeds a saturated counter-current flowed down the U-tube redirecting to the hot leg and the core outlet (see Figure 5).
1.3 THESIS PURPOSES

Royal Institute of Technology (KTH) participation in this benchmark consisted of the development of a new and independent TRACE (TRAC/RELAP Advanced Computational Engine) model of PWR PACTEL Test Facility based on SBL-50 experiment. A second simplified model was built in parallel to have a better understanding of several effects caused by the code, the model and user effect. The main purpose of the thesis is to analyze if TRACE code has the capability to simulate the experiment and discuss correct performance of vertical EPR type Steam Generators and Reactor Pressure Vessel under Natural Circulation conditions during Small-Break LOCA.

If the first goal is achieved, the next step is to carry out a space-time numerical analysis of the solution to verify its convergence. This work will provide relevant information to both code-users and developers about TRACE performance under Natural Circulation conditions with two-phase flow.
Chapter 2

Tools description

Advanced computing plays a fundamental role in the design, licensing and operation of NPP. As far as nuclear safety is concerned, it is important to be able to predict the response of the system to any possible perturbation. However, due to the huge complexity of systems and subsystems involved in a NPP, this goal is hard to accomplish without the use of additional tools. Scaled experimental test facilities is a good but expensive way of gaining experience and knowledge of the TH phenomena occurring during different accident scenarios, hence computer codes, which performance was already validated against earlier experiments, provide important information which contributes to make decisions concerning nuclear safety.

2.1 TRACE

TRACE [6] (TRAC/RELAP Advanced Computational Engine) is a computer code developed by the U.S. Nuclear Regulatory Commission (NRC) to model Thermal-Hydraulic behavior during postulated or operation accident conditions. This code was designed to unite and develop the capabilities of TRAC-P, TRAC-B and RELAP5.
• TRAC-P was able to study Large-Break LOCA in PWRs, while TRAC-B was used to analyze Large/Small Break LOCA in BWRs. Both codes had the capability to model TH phenomena in one-dimensional (1-D) and three-dimensional (3-D) components.

• RELAP5 (Excursion and Leak Analysis Program) is nowadays the main tool for analyzing Small-Break LOCA and transients in both PWR and BWR, being able to model thermal-hydraulic phenomena in one-dimensional volumes. Its long development has resulted in the design of an extremely consistent code, widely used around the world in safety analysis for design, licensing and operating NPPs.

Thus, TRACE is able to study large and small break LOCA and system transients in pressurized and boiling water reactors, modeling TH in both 1-D and 3-D space for more complex geometries and phenomena like the ones occurring in the vessel. Despite NRC describing TRACE code as “NRC's flagship thermal-hydraulic analysis tool”, and advising that “active maintenance of RELAP5 will be phased out in the next few years as usage of TRACE grows”, facts show that this transition from RELAP5 to TRACE is taking more than expected due to insufficient resources and efforts which are assigned to improve it. As a result, many code-users claim that TRACE still has many unresolved problems and needs further development. Nevertheless, TRACE is the code of the future due to its structure, simplicity and three-dimensional capabilities; therefore it needs to be validated against as many experiments as possible. For this reason, TRACE is the selected code to simulate Natural Circulation transient and investigate its behavior under such demanding conditions.

2.1.1 Code calculation methodology

TRACE code solves six field equations for two-phase flow: [7]

• Conservation of mass (liquid and vapor phase)
• Conservation of momentum (liquid and vapor phase)
• Conservation of energy (liquid and vapor phase)

Other additional equations can be solved for non-condensable gases (tracked with the vapor phase) and dissolved boron (tracked with the liquid phase) if necessary. To create a complete set of equations, the code uses semi-empirical correlations and equations such as:

• Equations of state (vapor and liquid phase pressures, temperatures and densities)
• Wall drag (pressure losses due to friction with pipes, area changes and pipe bends, etc.)
• Wall heat transfer (convection coefficient between the solid pipe and the fluids)
• Interfacial drag (force between liquid and vapor with different velocities)
• Interfacial heat transfer (energy transfer from vapor and liquid phases including mass transfer due to boiling or condensation)

The user must describe in the input file a discrete approximation of the system by designing a correct nodalization for the problem. Thermodynamic fluid state variables as temperature are calculated at the centre of each cell, while momentum equation variables like velocities are evaluated at the faces between two cells. By means of first-order and diffusive methods TRACE is able to calculate the following primary variables:

• Steam and non-condensable gas pressures
• Liquid and vapor temperatures and velocities
• Boron concentration
• Heat structure temperatures

The main components which are predefined in TRACE and used in this thesis are PIPE, TEE, FILL (mainly to control mass flow rate boundary condition and temperature), BREAK (to set pressure boundary conditions), VALVE, PUMP, HEAT STRUCTURE, POWER and TRIPS. These components are connected together by single junctions. Logical blocks and signals were also used to design specific controllers and obtain the desired output variables.

2.2 SNAP

SNAP [6] (Symbolic Nuclear Analysis Package) is a graphical user interface also developed by U.S.NRC which assists users in developing TRACE and RELAP5 models by automatically generating input files. The main advantage of this “CAD-style” software is its simplicity, which not only helps code-users to avoid typical errors in ASCII format but also makes the input file easier to read, understand and interpret. SNAP includes also post-processor tool which has the capability of creating animations which visually presents transient progression.

2.3 Additional software

• MATLAB is a prestigious and well-known high-technical computing language and interactive interface for algorithm design, data visualization and analysis. This software was selected as the main tool for extracting the data results from the output files TRACE generated (.xtv files usually treated with plotting software AptPlot); an existing algorithm was adapted to import the data into MATLAB’s workspace. Once the desired variables were available, a second algorithm was implemented to plot the results and make comparisons between models; thus, this
software is an efficient tool to analyze the simulation output. SNAP designers created a toolbox which allows users to extract TRACE outputs and data into MATLAB.

- Microsoft Excel was used to calculate and plot pressure losses along the Pressure Vessel and the loops. Comparisons between PWR PACTEL test facility results and TRACE outputs were performed with this calculation tool due to its simplicity.
Chapter 3

Models design

3.1 BASE MODELS DESIGN

After having introduced PWR PACTEL test facility and the simulation tools, this chapter is focused on procedures and criteria followed for models development.

3.1.1 Assumptions and simplifications

While modeling complex systems, e.g. NPP or test facility, one has to assume many reductions and simplifications. This subchapter will be focused on this problem.

- **1-D approximation**: exclusively one-dimensional nodalization is considered. This discretization, despite not fully representing reality, has been proved during decades of research to provide accepted results. Considering the geometry of scaled PWR PACTEL test facility, basically constructed with long pipes, trying to respect the heights and scaling the volumes, this conjecture should be correct.
**Null Heat losses:** despite PWR PACTEL heat losses are determined through rigorous characterizing experiments, it was decided not to model them since the efforts required were excessive compared to the small effect they have on the results. In general, test facilities and NPP are equipped with extremely efficient thermal insulation, so heat losses are usually insignificant. Thus, external pipes are modeled as perfect adiabatic heat-structures.

**Spatial – Plane adjustments:** 1-D approximation brings about some difficulties when trying to model pipe bends in the space plane. For instance, Hot and Cold Legs have multiple spatial bends to access SG or RPV inlets and these are not possible to model with a 1-D code. Nevertheless, since the effect of bends on TH behavior of fluids is important for pressure losses, the adopted solution was to create the tubes with the exact length and then introduce K-factor coefficients at the right places following the PWR PACTEL pressure losses (PL) experiments carried out to characterize the facility. Pressure losses play an important role specially if fluids encounter many transducers, bends, valves and area changes in the loop, thus, a special attention will be put on determining K-factor coefficients to model them (see section 3.1.3)

**Geometrical criteria:** during the models development, geometrical criteria design first consisted of creating simple pipes strictly respecting cross-flow area, elevations and length. Then, if non-acceptable discrepancies between component volumes from PWR PACTEL specification and TRACE model were found, strategic tubes which did not affect total elevation were slightly enlarged or shortened to respect as much as possible total volumes. For PWR PACTEL SG multi-tube exchanger and core, special attention to calculation of hydraulic diameter, cross flow area and modeling heat structure was made, since they are the one which “suffer” most from modeling simplifications.

**Symmetry:** PWR PACTEL test facility does not have completely symmetrical loops; as one of them includes a Pressurizer (PRZ) and a pressurizer line, its hot leg is slightly over-dimensioned in length, so in SS there’s actually a higher mass coolant flow through one of the loops than through the other one. These differences were removed in one of the models to check if they affected the results. Moreover, in SBL-50 PWR PACTEL experiment, FDW temperatures were also minimally asymmetric (23ºC versus 19ºC) which could cause convergence problems and originate unexpected and undesired behaviors; thus, this parameter was fixed to 23ºC for both secondary sides.
3.1.2 Models description

PWR PACTEL Test Facility elements were modeled in TRACE code using PIPE, TEE, SINGLE JUNCTION, VALVE components [8]. To simulate the inlet and outlet boundary conditions, TRACE components **FILL** and **BREAK** were used, respectively. The full nodalization diagram as SNAP graphical representation is presented in Figure 6:

![Figure 6: General view of PWR PACTEL TRACE model](image)
Other components are modeled as simple 1D pipe with precise dimensions and cross-sections obtained from PWR PACTEL General Description document. A cylindrical heat-structure with the right thickness, material properties and adiabatic boundary conditions was added for these simple tubes. That’s the case for the following list of components:

- Cold leg connector (1 pipe)
- Vessel downcomer (1 pipe)
- Lower plenum (5 pipes)
- Upper plenum (1 pipe)
- Hot Legs (2 pipes per loop)
- Cold legs (2 pipes per loop)
- Steam line (1 pipe per loop)

Nevertheless, the core and Steam Generators require a deeper analysis because they are much complex as it’s where the main heat transfers occur.

- **Core**

![Figure 7: Core TRACE nodalization](image)

The core is modeled as a single pipe (see Figure 7), but it contains two different heat-structures. The first heat structure is the external part of the pipe (walls), and it is modeled as an adiabatic cylinder. However, the active part of the core is represented by the second heat-structure which is defined by fuel rod option. This heat structure is a representation of all heating rods inside the core. The “Power Component” with the constant power option activated (155 kW), and correct axial and radial power shape is associated to this heat
structure. PWR PACTEL core disposes of three separated channels each one with a bypass and containing the same number of rods. Nevertheless, the core was modeled as a single channel with no bypass to simplify it.

- **Steam Generators (model 1)**

Steam Generators are more complex to model because is a multi-tube heat exchanger with many components, complicated geometry and heat fluxes. Therefore, code-user experience plays a fundamental role at this point. Sliced nodalization was imposed between U-tubes and boiler and downcomer to avoid heat transfer errors.

In the Model 1 the following solution was proposed (see Figure 8).

![Figure 8: Model 1 - Steam Generator view](image)

The part marked as “U-tubes” represents the primary side of SG. Its length is an average length of all U-tubes (about 6.5 meters, which is specified in *PWR PACTEL general description*) and the cross-section area is the sum of all U-tubes’ area (see the *Equation 1*). A scheme of heat exchange tubes in PWR PACTEL Steam Generators is presented in *Figure 9*. 
Figure 9: Heat exchange tubes of PWR PACTEL steam generators [3]

\[ A_p = n\pi \left(\phi_p - 2t\right)^2 / 4, \]

(Equation 1)

Where:

- \( n \) is total number of tubes (51),
- \( \phi_p \) is the external diameter of a single U-tube,
- \( t \) is the thickness of the tube.

The heat-structure is defined as a flat slab of thickness and length equivalent to the total wet-perimeter which can be calculated with the Equation 2. The reason why flat slab is chosen instead of cylinder is that the current TRACE version incorrectly uses Hydraulic Diameter in cylindrical heat-structures, and consequently, the heat transfer area is not correct causing mistakes in code predictions.

\[ P_w = n\pi(\phi_p - 2t) \]

(Equation 2)

Hence, all the U-tubes are lumped to a single pipe with the heat-structure, which is directly coupled to the boiler part of the SG. In the Figure 10 heat-structure connection between boiler of SG and U-tube components is presented. It is important to notice that when U-tubes reaches cell 9, the boiler outer surface keeps as cell 8 because top part of the U-tube has been reached, so the cold part of U-tube transfers energy to the same cell. After that, while U-tube cell number keeps increasing, boiler cells start decreasing until initial elevation is reached. Pipe 111 is the TRACE component number for U-tube pipe while Pipe 410 is the boiler’s component number.
The next component which needs attention is the SGs “boiler (pipe 410)”. It is a pipe of 6.5 meters length and sliced-nodalized with the U-tube node-elevation as a reference in order to correctly represent the heat-transfer. Its cross-sectional area is calculated as the secondary cross-section area minus the primary tubes area and minus the divisor plate area with the following equation:

\[ A_s = \frac{\pi (\phi_w^2)}{4} - n \frac{\pi (\phi_p - 2t)^2}{4} - \phi_w t_2 \]  

(Equation 3)

Where:

- \( \phi_w \) is the internal diameter of the wrapper.
- \( t_2 \) is the partition plate thickness in the secondary side.

This component has been modeled by three different heat-structures:

- First heat-structure is automatically generated when their cells were added as outer surface boundary conditions for U-tubes heat-structures.
- The other two correspond to the wrapper which separates the boiler (inner surface boundary condition) from each of two downcomers (outer surface boundary condition). Flat slab length is equivalent to the wrapper half-perimeter while its thickness is 4 mm, the
real value. Sliced nodalization is also applied between the cells of the boiler and the cell of both downcomers to minimize heat-transfer errors.

- **Steam Generators (model 2)**

Model 2 was designed according the same principles (primary and secondary cross-section, downcomer cross section and heat-transfer surface). The main difference appears in the way of representing SGs’ geometry (see Figure 11).

![Figure 11: Model 2 – Steam Generator view](image)

The boiler in this model is represented by two parts -hot and cold-. The hot part of U-tube heat structure is coupled to hot part of secondary side, while cold part of boiler is heated by U-tubes’ cold part. The feedwater is going only through one downcomer which feeds the hot side of the boiler, while internal circulation occurs with help of the second downcomer. These parts are separated by the divider plate which is modeled as a flat slab and has a thickness of 6mm and a length of 352.4 mm, equivalent to the diameter of the boiler and along the entire axial-direction of SG.

Another difference is the insertion of a single junction (SJ) to model possible cross-flow connections between hot and cold parts due to diffusion, velocity and density differences where the divider plate ends and the bending part of the U-tubes start. A TEE
component (which has the capability of introducing source terms to the 1-D motion equations to capture the influence of momentum from a side flow on the momentum in the main flow path) is also used to unify the mass flow rates from cold and hot parts to follow the path to the top part of the SG.

The heat exchange between both SGs downcomers and SGs’ boiler has been modeled with a heat-structure consisting of a flat slab. Its length is equivalent to the semi perimeter of the wrapper involving secondary side and its thickness is 4 mm as specified. Finally, another heat-structure is defined to model heat-transfer from both downcomers to the ambient air, imposing a null heat flux (adiabatic pipe) as boundary condition as assumed in section 3.1.

- **Controller for water level and pressure in SG**

The last step of SGs design was to develop a controller which maintains correct collapsed water level in the secondary side. A SNAP scheme of the controller’s logic is represented in the *Figure 12*; note that there are two equivalent controllers, one per steam generator; the controllers work the same way between both models.

<table>
<thead>
<tr>
<th>Level Controller SG I</th>
</tr>
</thead>
<tbody>
<tr>
<td>VOLLEV 100</td>
</tr>
<tr>
<td>DC Level</td>
</tr>
<tr>
<td>-101</td>
</tr>
<tr>
<td>Lag</td>
</tr>
<tr>
<td>DC Lagged Level</td>
</tr>
<tr>
<td>-102</td>
</tr>
<tr>
<td>PI</td>
</tr>
<tr>
<td>DC Level PI Signal</td>
</tr>
<tr>
<td>-100</td>
</tr>
<tr>
<td>Const</td>
</tr>
<tr>
<td>Target DC Level</td>
</tr>
</tbody>
</table>

*Figure 12: Collapsed level of steam generator controller*

*VOLLEV 100* is a signal of collapsed water level in the active part of the boiler. Control Block -101 (CB) adds a lag, while control block -100 is a constant, which is the desired water level. These two signals are the inputs to *CB -102*, a Proportional-Integral (PI) controller which provides an output signal error between both input parameters. The FDW mass flow rate is regulated by the error signal provided by PI controller.-

- **Steady-State controller for SGs**

Another controller (view *Figure 13*) was used to speed up the convergence to the correct temperature difference (ΔT) in U-tubes during steady-state calculations. A valve which regulates mass flow rate in the steam lines was added in order to reach the correct SGs performance (see *Figure 14*).
From signals TEMPF 200 and TEMPF201 which are temperatures of SG U-tube inlet and outlet respectively, the average value is calculated (CV 202). Next, a PI controller measures the error between the average value and the desired value (CV 203), and sends a feedback signal which regulates the opening valve area. This controller is a SS controller and is deactivated during the entire transient. It is important to remark that the values which regulate the feedwater pump and the valve are slightly different in both models as the Steam Generators have unequal designs.

---

**Pressurizer**

Since for SS calculations TRACE is treating Pressurizer (PRZ) components a BREAK component which maintain the desired pressure and water inventory in the primary side during steady-state calculations, the pressurizer was modeled as the BREAK component. This approach is possible since the pressurizer is isolated during entire transient and it does not influence results. In PWR PACTEL test facility the pressurizer was built in one of the loops, but in TRACE model corresponding BREAK component was placed in the top of upper plenum to resolve problems with steam accumulation during SS calculations.
Another problem connected to the Upper plenum, which was a “dead-ended” pipe, was thermal stratification obtained during SS. To help the code in obtaining correct temperature distribution in the upper plenum, it was necessary to add a FILL in its lower part which provides water with desired temperature during short time of calculation until the correct temperature distribution in the upper plenum is obtained. The excess of water in the system was drained by BREAK at the top of the Upper plenum (see Figure 15). This special treatment was necessary, because any fluid thermal stratification in the upper plenum will strongly influence the depressurization event during the transient.

![Figure 15: Upper plenum and pressurizer view](image)

- **Small-Break LOCA**

The SBLOCA is modeled with the help of BREAK component of 1mm of diameter and was located in the location indicated by benchmark specification, between the loop seal and the downcomer of RPV. An additional valve was added in the model which activates and deactivates LOCA at desired time. Figure 16 shows a schematic of the TRACE model for the Small-Break LOCA:
- **Trip for valve isolation**

  ![Figure 17: TRACE trip for valve isolation](image)

In this model there are several valves which need to be regulated during the entire transient. This regulation is done with the help of TRIP component (see Figure 17 and Figure 18). Appropriate trips were used for valves between SGs and steam lines, the valve between pressurizer line and PRZ and the valve between cold leg and LOCA Break.

![Figure 18: Logic of TRIP component](image)
Countercurrent flow limitation data model allows the user to apply correlations for CCFL by specifying locations and constants values to be used by the code. Three CCFL models were designed for junctions to the Steam Generators, Hot Legs inlet and Core inlet / outlet. Table 3 shows the constant values of these models, based on TRACE manual and experience from users [8] [9].

<table>
<thead>
<tr>
<th>CCFL MODELS</th>
<th>Bankoff interpolation</th>
<th>Slope</th>
<th>Correlation Constant</th>
</tr>
</thead>
<tbody>
<tr>
<td>Junctions to SGs</td>
<td>0</td>
<td>1</td>
<td>0.8</td>
</tr>
<tr>
<td>Hot legs inlet</td>
<td>0</td>
<td>1</td>
<td>0.35</td>
</tr>
<tr>
<td>Core inlet and outlet</td>
<td>0</td>
<td>1</td>
<td>0.8</td>
</tr>
</tbody>
</table>

Table 3: Countercurrent Flow Limitation data models

3.1.3 Determination of pressure losses

Researchers from PWR PACTEL performed several experiments to characterize the test facility; one of them was determination of single-phase pressure losses. RPV pressure losses had already been determined a decade ago during first experiments for VVER and horizontal SGs. Nevertheless, new constructed loops and vertical EPR type steam generators needed to be characterized.

An external low pressure emergency core cooling pump was connected to the attachment of the downcomer flow meter. The used procedure consisted of imposing a specific mass flow rate measured with an electromagnetic flow meter on the discharge side of the pump and recording data for 100 seconds. After that, a step increase of the mass flow rate was introduced and after 100 seconds of stabilization, pressure losses were acquired again for 100 seconds, and so on. In the Figure 19 it is shown a schematic representation of the transducers location along the downcomer, lower plenum, core and upper plenum and both legs of PWR PACTEL test facility:
The TRACE model was adapted to be able to measure pressure losses. PWR PACTEL experiments supplied pressure losses results for normal and reversal flow, but the model was tuned only for forward K-losses for the following reason. Hot legs and U-tubes where reverse flow is possible, especially during reflux mode, are simple long pipes. Therefore it was assumed that forward and reverse factors were the same value.

To determine pressure losses in the downcomer, lower plenum, core and upper plenum, other simplifications needed to be done to follow the experiment conditions. SGs were removed from the models; two fills were added in the CLs inlet, while two breaks were added on the HLs outlet (see Figure 20).
However, to determine pressure losses in hot legs, steam generators and cold legs from both loops, another solution for the model was done. Only secondary sides were removed and pump was added between the vessel downcomer and the lower plenum as it was carried out in the PWR PACTEL characterizing experiment (Figure 21). This pump was a controlled mass flow rate junction which allowed user to force the flow to the desired conditions.
The iterative procedure used to fit pressure losses of PWR PACTEL test facility by adapting appropriate K-factors in TRACE is explained in details next.

Pressure Losses were calculated by isolating the $\Delta P$ term in Bernoulli’s principle:

$$\frac{P_1}{\gamma} + \frac{c_1^2}{2g} + z_1 = \frac{P_2}{\gamma} + \frac{c_2^2}{2g} + z_2 + \Delta P$$  \hspace{1cm} (Equation 4)

The terms $g$ and $\gamma$ were considered as constants, and the heights ($z$) were measured in the model by checking the component geometry. By running the calculations, it was possible to obtain the results in steady state calculation for pressure and fluid velocity, allowing to determinate pressure losses ($\Delta P$). This term is dependent on several factors such as fluid velocity, flow regime and K-factor of the pipe (which depend on the component geometry). The flow area cannot be changed and the mass flow rate was imposed in the experiment, so the fluid velocity was fixed due to the continuity equation:

$$Q_1 = c_1A = Q_2 = c_2A$$  \hspace{1cm} (Equation 5)

Consequently, the only possibility to fit the results was by guessing the correct K-factors of all the pipes in the vessel and loops, running the calculation and comparing the results with the given ones in PWR PACTEL’s experiment. This parameter was defined in the input file in edges of every cell (where velocities are calculated). Microsoft Excel was used to create a calculation sheet where comparison graphs were automatically plotted. Annex 1 contains the graphs comparisons, but as an example Figures 22 - 24 present some results:
Figure 23: Pressure difference [Pa] VS mass flow rate [kg/s] comparison in D0074 (Hot Leg I)

Figure 24: Pressure difference [Pa] VS mass flow rate [kg/s] comparison in D0086 (U-tube SG1)
Once pressure losses were correctly predicted in both TRACE models, the base models were finally complete and they were used without further modifications for the prediction of the SB-LOCA experiment. Results will be shown together with the rest of support models in the following chapter, where they will be discussed and analyzed.

3.2 SUPPORT MODELS CONSTRUCTION FOR TIME-SPACE ANALYSIS

Since one of the thesis objectives was achieved - the TRACE model for PWR PACTEL test facility had been successfully designed and validated for steady-state -, the next step can be performed. The next step is to carry out the transient simulations including space and time convergence study. During renodalization of the models, SNAP automatically lumps K-factors values trying to conserve the previous behavior.

3.2.1 Renodalization

Four different nodalizations for each of the two TRACE models were developed to carry out this analysis. The main changes in the nodalizations will be described and explained in detail. Choosing a correct nodalization for this problem is not an easy task, as results of a system codes such as TRACE are mesh-dependent. The best nodalization will be the one which has the ability to accurately predict all TH phenomena occurring during SS and transient in a reasonable computational time.

To be able to make comparisons between both models, exactly the same RPV was introduced in TRACE models, with the same number of nodes and edge location. Moreover, despite geometrical and design differences in SGs between both models (one of them including cross-flow and flat slab dividing hot and cold parts), same number of vertical nodes was imposed for U-tubes in all cases. Applying these measures, errors were minimized when comparing efficiency and accuracy between models to predict SBL-50 experiment. Figures of the different models can be found in the annexes, but as the illustrative examples, SNAP graphical representation of SGs for both models will be shown to get an idea of important differences.
- **Coarse nodalization (G):** From the base models, the renodalization consisted in meshing nodes all along the primary and secondary side in order to obtain node-lengths of about 0.5 meters in SG and about 0.6 meters in the power part of the core (see Figure 25). Following TRACE user’s manual recommendations [8] [9], this length should be good enough for those parts where properties don’t experience many changes (as all through hot and cold legs or the downcomer of RPV), but it could be inefficient to predict TH phenomena occurring in the Core and in U-tubes of Steam Generators where heat transfer plays a fundamental role and nodal diffusion errors connected with the course grid can have an impact to the results. Knowing that TRACE changes elevations of re-meshed non-vertical nodes, special attention was paid to not modify these parameters as they could substantially alter loop elevations. Nevertheless, this model was designed inaccurately on purpose, to check if user-effects caused by lack of experience could significantly affect the results.

![Figure 25: Steam Generators - coarse nodalization models](image-url)
- **Intermediate nodalization (M):** Node-size modifications from base-models were carried to reach an average node-length of 30 cm in SGs and around 50 cm in the active part of core (*Figure 26*). The downcomer, lower plenum, upper plenum, CLs and HLs were renodalized and refined by two, getting a more accurate model. Special attention was paid to not modify elevations when activating the renodalization was done.

*Figure 26: Steam Generators intermediate nodalization models*

- **Fine nodalization (P):** This fine nodalization followed TRACE user’s manual recommendations and experience from code-users for meshing the model components. Thus, a node length of 24 cm was chosen to mesh the core, while a finer nodalization was applied to both SGs, having a length slightly over 16 cm. In theory, results should be considerably improved with this mesh, as fluid state-variables, momentums and their changes are precisely calculated. Measurement errors due to location difference of real transducers and PWR PACTEL test facility and code calculation points should also be minimal. As it’s possible to appreciate in the *Figure 27*, both steam generators have now fine and precise nodalization, which also causes an increase of calculation time.
- **Fine-sliced nodalization (S):** This fourth nodalization is the most elaborated PWR PACTEL test facility nodalization that was developed. The concept of slice-nodalization comes from a strategy which attempts to minimize the error propagation due to node asymmetries along the fluid flow path. Thus, this model has completely regular sliced-cuts through the entire model, which accurately create a very fine nodalization. The fundamental components to select for the “cutting” elevations were the lower plenum for lowest parts of the facility, the core for mid-elevation parts and SGs for the top parts of the facility. So these elements remained constant compared to the previously designed “fine-nodalization”, and the rest of components (downcomer, upper plenum and cold legs were modified to fulfill the slice-criteria. Figures of the SG are not shown, because they are exactly the same as in the previous model (model P). It is pertinent to remember that “complex model” doesn’t necessarily mean “best model” or “accurate model”.

*Table 4* presents number of hydraulic volumes or cells and its location (reactor pressure vessel, loops or secondary side). Number of nodes increases with level of model refinement. In the *Annex 2* it is possible to find the SNAP graphical representation of all the submitted models.
### 3.2.2 Timestep data

Once the models’ space-mesh is fully defined, it is necessary to analyze time step influence on the solution, convergence of SS and numerical stability. The solution would not be accepted if it is sensitive to time step. The problem time can be divided in separate time domains (e.g. blowdown, reflood), where time step properties can be changed, if necessary. For instance, when SS is close to being reached, a higher maximum time step size can be more efficient, while lower values should be used during demanding transients, when changes occur fast and high accuracy provides more realistic results. By adjusting these parameters the overall calculation time can be considerably reduced, obtaining a more efficient performance. Nevertheless, a single time interval from the beginning to the end of transient was selected to study the whole problem. The most relevant time step data options are briefly described next (see also Figure 28).

- **End time (TEND):** It represents the end time for the specific time domain. For the studied problem, this value was selected to 25 000 seconds. The LOCA break valve opens at 10 000 seconds, so 15 000 seconds of transient are calculated in every simulation. It's pertinent to remember that PWR PACTEL experiment was
stopped when core dry out occurred and temperatures in the core increased over the pre-specified limits.

- **Minimum size (DTMIN):** Minimum size was assigned the possible lowest value (1E-20s) to avoid errors due to time step reduction. However, it’s necessary to explain that DTMAX also plays an important role in time step selection algorithm; the code cannot reduce time step size more than a fixed value, so if DTMAX is too big, the time step reduction error can occur even if DTMIN would allow such time step to be used.

- **Maximum size (DTMAX):** DTMAX is one of the most relevant parameters, because it determines the maximum time step which can be used in a calculation during a defined time interval. High DTMAX values reduce calculation time and simulation will be completed faster, while low values increase calculation time. As calculation time is an important factor to consider, interesting conclusions can be reached by changing DTMAX value and comparing time convergence. In this work, three different DTMAX values (100ms / 10ms / 1ms) were selected to contrast the results and time step dependence of the solution.

### 3.3 MAIN PARAMETERS

The model performance comparison is based on the most relevant parameters and variables from the primary side, since the secondary side remains almost constant during all the analysis. Comparison graphs plotting the evolution of the following parameters during transient are the main tool to compare the different models, nodalizations, time-steps and results accuracy.

- Vessel downcomer mass flow rate
- Cold Legs I / II mass flow rate
- Core inlet temperature
- Core channel outlet temperature
- SG I / II inlet temperature
- SG I / II outlet temperature
- Upper plenum bottom / top temperature
- Upper plenum top pressure
- Differential Pressures
- Collapsed level downcomer
- Collapsed level Pressure Vessel
- Dry out time
- Integrated Mass leaked out
- ΔT in core
- ΔT in SG I / II
Chapter 4

Results analysis

4.1 MODEL VALIDATION

The calculation time comparison (in hours) between model 1 support nodalizations and different maximum time-steps is presented in Table 5. All these calculations were submitted in the same computer and 15,000 seconds transient was initiated after 10,000 seconds of Steady State (SS) calculation. Begin of Transient (BOT) starts at time 10,000 seconds.

<table>
<thead>
<tr>
<th>CALCULATION TIME</th>
<th>MAXIMUM TIMESTEP DATA</th>
</tr>
</thead>
<tbody>
<tr>
<td>NODALIZATION</td>
<td>100ms</td>
</tr>
<tr>
<td>Coarse</td>
<td>2.5h</td>
</tr>
<tr>
<td>Intermedium</td>
<td>2.75h</td>
</tr>
<tr>
<td>Fine</td>
<td>3h</td>
</tr>
<tr>
<td>Fine-sliced</td>
<td>3h</td>
</tr>
</tbody>
</table>

Table 5: Calculation time for Model 1 support nodalizations
It is possible to appreciate how reducing maximum time step causes a remarkable increase of calculation time (for fine-sliced calculation 3 hours for 100ms versus more than one week if 1ms maximum time step is used). A refined nodalization also increases calculating time as TRACE has to solve the two-phase flow and heat transfer equations in a larger number of nodes, but its effect are not as significant. This parameter must be taken into consideration when deciding the best models.

Steady State converged values of main parameters for Model 1 and Model 2 and their comparison with PWR PACTEL’s experiment results are presented in Table 6. Relative errors are also calculated in the last two columns.

<table>
<thead>
<tr>
<th>Code</th>
<th>PARAMETER</th>
<th>UNITS</th>
<th>SS M1 VALUE</th>
<th>SS M2 VALUE</th>
<th>SBL-50 VALUE</th>
<th>Error M1 (%)</th>
<th>Error M2 (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Mass flow rate downcomer</td>
<td>kg/s</td>
<td>1.4174</td>
<td>1.3577</td>
<td>1.1867</td>
<td>19.44%</td>
<td>14.41%</td>
</tr>
<tr>
<td>2</td>
<td>Mass flow rate Cold Leg 1</td>
<td>kg/s</td>
<td>0.727</td>
<td>0.6839</td>
<td>0.6359</td>
<td>14.33%</td>
<td>7.55%</td>
</tr>
<tr>
<td>3</td>
<td>Mass flow rate Cold Leg 2</td>
<td>kg/s</td>
<td>0.6904</td>
<td>0.6738</td>
<td>0.5968</td>
<td>15.68%</td>
<td>12.90%</td>
</tr>
<tr>
<td>4</td>
<td>Temperature Core Inlet</td>
<td>K</td>
<td>527.17</td>
<td>525.46</td>
<td>525.37</td>
<td>0.34%</td>
<td>0.02%</td>
</tr>
<tr>
<td>5</td>
<td>Temperature Core Outlet</td>
<td>K</td>
<td>549.03</td>
<td>548.42</td>
<td>549.52</td>
<td>0.09%</td>
<td>0.20%</td>
</tr>
<tr>
<td>6</td>
<td>Temperature Hot Leg 1 Outlet</td>
<td>K</td>
<td>549.03</td>
<td>548.47</td>
<td>549.68</td>
<td>0.12%</td>
<td>0.22%</td>
</tr>
<tr>
<td>7</td>
<td>Temperature Cold Leg 1 Inlet</td>
<td>K</td>
<td>527.18</td>
<td>525.28</td>
<td>527.13</td>
<td>0.01%</td>
<td>0.35%</td>
</tr>
<tr>
<td>8</td>
<td>Temperature Hot Leg 2 Outlet</td>
<td>K</td>
<td>549.03</td>
<td>548.48</td>
<td>548.22</td>
<td>0.15%</td>
<td>0.05%</td>
</tr>
<tr>
<td>9</td>
<td>Temperature Cold Leg 2 Inlet</td>
<td>K</td>
<td>527.14</td>
<td>525.28</td>
<td>526.52</td>
<td>0.12%</td>
<td>0.24%</td>
</tr>
<tr>
<td>10</td>
<td>Temperature Upper plenum (Top)</td>
<td>K</td>
<td>549.03</td>
<td>548.46</td>
<td>543.69</td>
<td>0.40%</td>
<td>0.01%</td>
</tr>
<tr>
<td>11</td>
<td>Temperature Upper plenum (Bottom)</td>
<td>K</td>
<td>547.16</td>
<td>545.07</td>
<td>545</td>
<td>0.98%</td>
<td>0.88%</td>
</tr>
<tr>
<td>12</td>
<td>Pressure Upper plenum</td>
<td>bar</td>
<td>75.02</td>
<td>74.03</td>
<td>74.537</td>
<td>0.65%</td>
<td>0.68%</td>
</tr>
<tr>
<td>13</td>
<td>Pressure Steam Generator 1</td>
<td>bar</td>
<td>42.01</td>
<td>42.01</td>
<td>41.659</td>
<td>0.84%</td>
<td>0.84%</td>
</tr>
<tr>
<td>14</td>
<td>Pressure Steam Generator 2</td>
<td>bar</td>
<td>42.01</td>
<td>42.01</td>
<td>41.935</td>
<td>0.18%</td>
<td>0.18%</td>
</tr>
<tr>
<td>15</td>
<td>Differential Pressure Hot Leg 1</td>
<td>bar</td>
<td>19278</td>
<td>19404</td>
<td>19909</td>
<td>3.17%</td>
<td>2.54%</td>
</tr>
<tr>
<td>16</td>
<td>Differential Pressure Cold Leg 1</td>
<td>bar</td>
<td>19794</td>
<td>19910</td>
<td>20820</td>
<td>4.93%</td>
<td>4.37%</td>
</tr>
<tr>
<td>17</td>
<td>Differential Pressure Hot Leg 2</td>
<td>bar</td>
<td>19313</td>
<td>19400</td>
<td>19708</td>
<td>2.00%</td>
<td>1.56%</td>
</tr>
<tr>
<td>18</td>
<td>Differential Pressure Cold Leg 2</td>
<td>bar</td>
<td>19780</td>
<td>19904</td>
<td>21077</td>
<td>6.15%</td>
<td>5.57%</td>
</tr>
<tr>
<td>19</td>
<td>Collapsed level downcomer</td>
<td>m</td>
<td>7.86</td>
<td>7.86</td>
<td>7.78</td>
<td>1.03%</td>
<td>1.03%</td>
</tr>
<tr>
<td>20</td>
<td>Collapsed level upper plenum</td>
<td>m</td>
<td>12.39</td>
<td>12.37</td>
<td>12.25</td>
<td>1.14%</td>
<td>0.98%</td>
</tr>
<tr>
<td>21</td>
<td>Differential temperature in Core</td>
<td>K</td>
<td>21.86</td>
<td>22.96</td>
<td>24.15</td>
<td>9.48%</td>
<td>4.93%</td>
</tr>
<tr>
<td>22</td>
<td>Differential temperature in SG1</td>
<td>K</td>
<td>21.84</td>
<td>23.19</td>
<td>22.55</td>
<td>3.13%</td>
<td>2.84%</td>
</tr>
<tr>
<td>23</td>
<td>Differential temperature in SG2</td>
<td>K</td>
<td>21.88</td>
<td>23.20</td>
<td>21.70</td>
<td>0.84%</td>
<td>6.91%</td>
</tr>
</tbody>
</table>

Table 6: Models VS PWR PACTEL SBL-50 experiment steady state comparison
TRACE Steady State results show considerable discrepancies in some parameters, mainly in total mass flow rate, differential temperature in core and differential pressure in cold legs which need to be justified:

- The differential pressure can be caused by uncertainty in the experiment results (the measurement error by transducers was relatively important in the characterization experiments) as well as by user-effect in K-factors determination, which were obtained for different conditions than the SBL-50 test itself.

- Error in mass flow rate downcomer can be explained by several factors. First of all, small volume differences, heat-transfer areas and elevations can considerably affect the natural circulation mass flow rate; thus, user-effect errors are amplified in NC problems. Mass flow rate measurement in SBL-50 also has a higher uncertainty than other kind of measurements (25%), so TRACE value is still inside the margins. The lack of accuracy in total mass flow rate prediction brings about secondary errors; for instance, mass flow rate in cold legs will also experience similar discrepancies.

- The differential temperature error in the core and in the SGs is also a consequence of the total mass flow rate mismatch. In PWR PACTEL facility, the ΔT in the core is higher than in both TRACE models because total mass flow rate is lower, consequently experiencing a larger temperature increase for the same heat power input.

Nevertheless, TRACE model is considered as good enough, as general behavior of transient’s phenomena (temperatures, core and SG performance, depressurizations, reflux mode, is accurately predicted (see Figures 29 – 35).

Figure 29: Model comparison upper plenum top pressure
Figure 29 shows how first depressurization and Plateau are correctly predicted from BOT until 3000 seconds after BOT, but both models suffer the second depressurization 500 seconds earlier than in PWR PACTEL experiment. User-effect can explain these differences; for instance, K-factors in upper plenum were not calculated, and small cross-sectional differences may cause water level to decrease faster. Moreover, an existing diffuser in PWR PACTEL facility which mixed the mixture could also affect the data obtainment by instruments and transducers. This error is also reflected in Figures 30, 31 and 32, where differential temperatures in the core and in both SGs are correctly predicted except when they follow the anticipated depressurization around 3000 seconds after BOT. The different behavior between Model 1 and Model 2 just before BOT is caused by the valve controller, which causes small oscillations in one of them. Nevertheless, the discrepancy, which follows a realistic behavior, is small enough and it is still inside the error margins.
Integrated mass leaked out from the facility through the break is represented in Figure 33, in which both models experience similar behavior and they fit PWR PACTEL’s results well.

Figures 34, 35 plotting differential pressures in hot and cold leg also show that models are validated for this experiment as it is able to predict its behavior. K-factors values and the transient evolution are in general well estimated.
Figure 34: Model comparison differential pressure Hot Leg I

Figure 35: Model comparison differential pressure Cold Leg I

Figure 36 shows total mass flow rate entering the reactor pressure vessel from downcomer during the transient. Both models have a maximum mass flow rate overestimation error around 50% and it also occurs earlier than expected. Steady State mass flow error can explain part of this error, as when the valve break opens the system accelerates and slows down in a higher rate. The experimental values uncertainty can also cause this large difference. Nevertheless, PACTEL test facility researchers admitted the presence of non-condensable gases accumulated at the top of U-tubes which could limit the primary mass flow rate.
Another difference between the two models and PWR PACTEL results is shown in Figure 37, which is the collapsed level of water in the vessel downcomer. While in Model 1, the super-heated steam starts accumulating at the top of the cold leg connector producing a decrease in the water level, Model 2 correctly predicts the parameter. The reason why this occurs is the difference in Steady-State convergence values between both models. Primary side temperature is some degrees higher in the first model because of the different design of SGs; consequently, just after LOCA break opens and saturation conditions are reached, a small quantity of steam remains trapped at this top part (BOT). The water level remains approximately constant until second depressurization occurs (3,000 seconds after BOT) when the steam gets stuck again and altering the collapsed water level signal. However, this difference doesn’t affect much the rest of the parameter results because steam never reaches the downcomer inlet.
Finally, the most relevant difference appears when plotting mixture temperature in the core channel outlet in order to study the dry-out in the core (see Figure 38). While Model 2 is able to predict core dry-out at the same time as PWR PACTEL experiment (approximately 9,700 seconds after BOT) results from Model 1 show this phenomenon about 1000 seconds later than expected. As it is possible to see in Figure 39, the final decrease of pressure vessel collapsed level occurs at the same time in Model 2 and PWR PACTEL experiment (8,700 seconds after BOT), while it takes longer in Model 1. Considering that geometrical differences are minimal in the vessel and the legs, the explanation for this difference lies in the Steam Generators nodalization. In model 1 the Natural Circulation establishes a higher mass flow rate than model 2, and as the differential temperature is similar in both models, a higher power is removed through them. Hence, one of the conclusions we can obtain from this comparison is that Model 2 offers a much more realistic representation of PWR PACTEL test facility.

![Figure 38: Model comparison core channel outlet temperature (dry-out)](image1)

![Figure 39: Model comparison of collapsed level of the Reactor Pressure Vessel](image2)
4.2 DISCUSSION OF TIME-SPACE CONVERGENCE

4.2.1 Time step effect

Time step sensitivity was carried out by running three different calculations with the same model and only changing the maximum time step. In general, the results didn't change much at any of the models, which is a positive aspect as it reflects that the solution is time-convergent (results could not be trusted if by changing this parameter they changed too much) - see Figure 40.

![Differential Temperature Steam Generator II](image1)

Figure 40: Maximum time step comparison. Model 2 - Differential temperature in SG II

In Figure 41 it is possible to appreciate an earlier second depressurization between calculations performed with 100ms $DTMADX$ and the rest. Nevertheless, the results between the other simulations are quasi-identical.

![Model 1 - Upper plenum (top) Pressure](image2)

Figure 41: Maximum time step comparison. Model 1 – Pressure in top upper plenum
However, with the largest maximum time step calculation (100 milliseconds) the mixture temperature at the core outlet has different behavior when dry-out occurs. Even if experimental data is not available – PWR PACTEL SBL-50 experiment ended before damage to the facility - and TRACE code is not as accurate under these conditions, this observed result is not desirable because the discrepancies are significant – see Figure 42.

![Figure 42: Maximum time step comparison. Model 2 – Core channel outlet temperature](image)

It is important to notice that the coarser the nodalization, the more influence the maximum time step has on the results. In fine-sliced nodalization no relevant differences could be appreciated, while in the coarse model the changes in the calculations were more noticeable. However, dry-out evolution and peak temperature can be underestimated if a large maximum time step is chosen, so the recommendation for would be to use at most 10ms for DTMAX, as it is demonstrated that the solution has time-converged.

### 4.2.2 Nodalization comparison

Nodalization comparison is an important tool to study numerical sensitivity of the solution; if the outputs experience significant changes by nodes refinement, the model would not be reliable as it would not be time-convergent. In the next figures, graphical comparisons are shown for the different nodalizations - coarse (G), medium (M), fine (P) and fine-sliced (S) - and PWR PACTEL data are presented and discussed. Figure 43 shows how coarse nodalization experiences slightly different behavior compared to the other nodalizations. Steady state value for ΔT in steam generator II convergences to a different value and is not able to follow correctly Plateau – from 200 seconds until 2,700 seconds after BOT due to oscillations. After second depressurization (3,000 seconds after BOT) its behavior improves. It can be concluded that using too coarse nodalization affects TRACE calculation results and the solution has not numerically space-converged. Nevertheless,
intermediate nodalization is already space-converged as it hardly reveals discrepancies with fine and fine-sliced nodalization results.

Figure 43: Nodalization comparison. Model 1 – Differential temperature in SG II

Figure 44 shows another possible error caused by the use of inappropriate nodalization. Even if the evolution of the phenomena during the transient is well-represented, the final pressure stabilizes at a considerably higher value than the other models. This pressure error causes different behavior and transient evolution, as it is possible to observe in Figure 45, where all the models are able to predict (even if a bit later than it should be) a core dry-out except from coarse nodalization model.

Figure 44: Nodalization comparison. Model 1 Upper plenum (top) pressure
Another specific conclusion that can be observed is that the difference between fine and fine-sliced nodalization for this SBLOCA in natural circulation conditions is practically nonexistent. If the nodalization is fine-enough, sliced nodalizations do not seem to show any improvements in the results. Hence, models with refinement level of 30 centimeters per node in Steam Generators and 50 centimeters per node in the core are able to correctly predict this type of transient.

Figure 45: Nodalization comparison. Model 1 - Core channel outlet temperature
Chapter 5

Conclusions

The main purpose of this project was achieved, as a validated TRACE representation of PWR PACTEL test facility and simulation of SBL-50 experiment has been successful. Vertical European Pressurized Water Reactor type steam generators performance was correctly predicted during an SBLOCA in Natural circulation. Both biphasic and reflux mode cooling stages can be visualized and main physical and TH phenomena can be explained.

TRACE time and space convergence revealed that the proposed models give numerically converged solutions and it is fully reliable and correct. Time analysis shows that in DTMAX maximum time step should not be higher than 10ms as the code can miss important events such as core dry-out and results accuracy are clearly affected. Space studies show that TRACE manual recommendations provide correct and converged results, and even intermediate nodalization already provided satisfactory results. Nevertheless, if too coarse nodalization is used, predictions can be highly erroneous, so TRACE user's guide should be carefully followed.

Considering calculation time, the best model to study SBLOCA transients under NC conditions is the medium nodalization with 10ms maximum time step, as it is able to predict all TH phenomena involved in the scenario in a reasonable time. The experience gained and developed during this work is going to be useful for both users and developers to keep improving the code and complete the evolution-step from RELAP5 to TRACE.

Further studies related to this thesis should be directed to study sensitivity of other parameters on the model’s results, like CCFL models constants. TRACE code needs to keep developing and feedback between developers and users must keep on by validating the code with more experimental tests, and a lot of research is still to be done in this field.
References


Annexes

A1) Pressure Losses comparison

A2) Models representation
Annex 1: Pressure losses comparison

This annex shows the graph comparisons between PWR PACTEL test facility experimental pressure losses and TRACE model losses. In the model, K-factors were guessed during several iterations until all the calculated points were satisfactory fit. The first five plots represent differential pressure in the reactor vessel.

A1: Figure 1. PWR PACTEL test facility’s differential pressure transducers location in the vessel [3]
A1: Figure 2. Differential pressure from lower plenum outlet to active core inlet

A1: Figure 3. Differential pressure from downcomer outlet to lowest point of lower plenum
A1: Figure 4. Differential pressure from lowest point of lower plenum to lower plenum outlet

A1: Figure 5. Differential pressure in active part of core
A1: Figure 6. Differential pressure from cold leg connector to lowest point of lower plenum
The next plots represent forward differential pressure losses comparison along first loop (hot leg, steam generator I and cold leg).

A1: Figure 7. PWR PACTEL test facility’s differential pressure transducer location in the loops [3]
A1: Figure 8. Differential pressure in Hot Leg I

A1: Figure 9. Differential pressure in Steam Generator I
A1: Figure 10. Differential pressure from outlet Steam Generator I to lowest point of Cold Leg I

A1: Figure 11. Differential pressure lowest point of Cold Leg I to Cold leg connector
Finally, the last five plots represent forward differential pressure losses along second loop (hot leg, steam generator II and cold leg II).
A1: Figure 14. Differential pressure in Steam Generator II

A1: Figure 15. Differential pressure from outlet Steam Generator II to lowest point of Cold Leg II
A1: Figure 16. Differential pressure lowest point of Cold Leg II to Cold leg connector

A1: Figure 17. Differential Pressure in the U-tubes of Steam Generator II
Annex 2: Models representation

A2: Figure 1. Model 1 coarse nodalization

A2: Figure 2. Model 1 intermedium nodalization
A2: Figure 3. Model 1 fine nodalization

A2: Figure 4. Model 1 fine-sliced nodalization
A2: Figure 5. Model 2 coarse nodalization

A2: Figure 6. Model 2 intermedium nodalization
A2: Figure 7. Model 2 fine nodalization

A2: Figure 8. Model 2 fine-sliced nodalization