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Thermal-hydraulic code assessment activities of the Technical University of Catalonia

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Abstract

For the past 15 years, the Technical University of Catalonia (UPC) has been involved in analysis and support to exploitation tasks related to the thermal-hydraulic behavior of commercial nuclear power plants. There has been a close cooperation with 3 Catalan NPPs (Ascó-I and II and Vandellòs-II), all of them 3-loop PWRs of Westinghouse design. More recently, the research activities of the thermal-hydraulic group of the UPC have diversified, including subjects like advanced reactors passive safety and 3-D TH coupled neutronics.

The UPC team analyzed (using RELAP5/Mod3.2) some PANDA experiments in relation to the TEPSS project (EU) and ISP-42. PANDA is an integral test facility to study, among other issues, the interaction between containment and primary system and the performance of passive safety features. One of the conclusions that came out from those benchmarks was that the mixing phenomena taking place in the containment couldn't be simulated with a 1D model, but a system code like RELAP5 was necessary, considering the complex interaction between reactor and containment systems. Although some progress has been achieved by using pseudo-3D models (parallel 1D components linked by cross-flow junctions) there is still room for a full 3D system code to improve results.

Our team has also participated in the OECD-MSLB benchmark, in a partnership with Pisa University (Italy). A hypothetical MSLB scenario was assumed for the TMI-I power plant, and UPC-UP performed the calculation using either QUABOX and PARCS coupled to RELAP5. From the experience gained in that benchmark, a PARCS-RELAP5 coupled neutronic-TH 3D model was developed for Ascó-I power plant. Limitations arose when trying to validate the thermal-hydraulic component of the model, as it is, actually, a pseudo-3D model. So, there is also in this subject some room for improvement by using a 3D code.

This paper gives some details on the aforementioned assessment activities as well as on future issues that are planed in the field of commercial plant modeling. The UPC's interest in having a suitable 3D tool to increase its capabilities for the assessment of safety related issues of present and future nuclear power plants will be explained.

Our proposal to perform some validation calculations taking advantage of the experience acquired on commercial power plants, is also presented. A transient occurred at Vandellòs-II, in which the loss of offsite power produced the trip of the reactor, the turbine and the RCPs, has been selected for this purpose. It is a fully asymmetrical transient in which it will be interesting to follow the behavior of the loops during a restart sequence of

the pumps using RELAP5-3D and to compare these results with plant data and with the prediction obtained with a RELAP5/Mod3.2 pseudo-3D model of the plant.

INTRODUCTION

The thermal-hydraulic (TH) group of the Technical University of Catalonia (UPC) was created about 15 years ago with the aim of modeling and simulating the behavior of integral PWR power plants in operational as well as in accidental scenarios.

During this period the UPC group has been closely cooperating with the three PWR nuclear power plants sited in Catalonia. These plants are Ascó-I, Ascó-II and Vandellòs-II, all of them three-looped Westinghouse designs of about 1000 MWe.

The tasks performed by the UPC in the framework of this cooperation have consisted of different kinds of thermal-hydraulic analyses, either related to requirements of the Nuclear Safety Council (the Spanish equivalent of the USNRC) or to give support to exploitation engineering. The main tool of the UPC thermal-hydraulic group has been the RELAP5 code in the different versions released during these years.

Analyzing and investigating phenomena of accidental scenarios has allowed our group to better understand the behavior of these plants and to improve their models. Besides, the analyses done during these years have also been useful to have an insight into the code, either to validate the build-in RELAP5 models or to detect some possible anomalies [1].

Among the last activities performed for the commercial power plants, the documentation, validation and qualification of their models has been one of the most relevant [2].

Since year 1995, the research activities of the thermal-hydraulics research group of the UPC have diversified. The areas covered more or less intensively include severe core damage, reactor control simulation and, related with code assessment, subjects like advanced reactors passive safety and 3-D thermal-hydraulics coupled neutronics.

PARTICIPATION IN PANDA EXPERIMENTS ANALYSES

In 1995 the UPC TH group began to cooperate with General Electric, in the framework of the ESBWR project. Although this collaboration is presently in a state of stand-by, the thermal-hydraulics group of the UPC participated in the ESBWR related TEPSS project, partly funded by the Commission of the European Communities within the Fourth Framework Program.

TEPSS (Technology Enhancement for Passive Safety Systems) project, which lasted from 1996 until 1998, had the main objective of making significant contributions to the base technology of Passive Advanced Boiling Water Reactors. With this aim, experimental work and analytical research was undertaken in different areas, one of them being the study of the residual heat removal by passive means after a loss of coolant accident into a simplified boiling water reactor type containment. Experiments in this area were performed in the PANDA facility at the Paul Scherrer Institute (Switzerland) [3].

PANDA is an integral test facility to study, among other issues, the interaction between containment and primary system and the performance of passive safety features. PANDA consists of large interconnected vessels (with a typical diameter of 4 meter) that simulate the containment of a passive BWR (figure 1). The facility incorporates three condensers simulating the Passive Containment Cooling System (PCCS) and optionally a fourth condenser simulating the Isolation Condenser. A tall vessel simulates the reactor pressure vessel (RPV) and provides steam produced by electrical heaters (the installed power is 1.5 MW). The facility is instrumented with over 500 sensors. Axial distributions of temperatures and non-condensable concentrations in the vessels can be obtained from thermocouple, pressure and a limited number of air probe measurements.

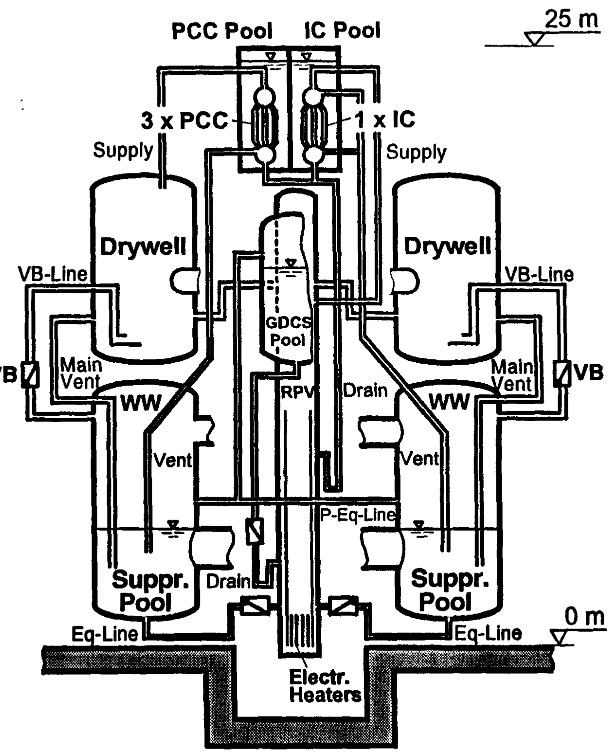

Figure 1: *Drawing of the PANDA facility*

The UPC provided analytical support of two of the eight PANDA experiments[1] within TEPSS project, performing either pre-test and post-test calculations using RELAP5/Mod3.2 [4].

Also in relation with the PANDA facility, the UPC participated in the International Standard Problem Exercise ISP-42. This test was run on April 1998 under the auspices of the OCDE/NEA Committee for the Safety of Nuclear Installations (CSNI) and was financially supported by the Research Foundation of the Swiss Utilities (PSEL).

The main aims of the ISP-42 were to gain insights into passive heat removal systems, to assess the ability of the models implemented in the codes with respect to the physical phenomena of interest (low pressure, low driving force) and to test the capabilities of internationally used thermal-hydraulic codes to analyze passive decay heat removal systems.

One of the conclusions that came out from those benchmarks was that the complex mixing/stratification phenomena that takes place in the containment (involving pure steam at distinct temperatures as well as air/steam mixtures in different proportions) couldn't be simulated correctly with a simple 1D model. Nevertheless a system code was proved to be necessary, considering the complex interaction between reactor and containment systems. Although some progress in the simulation of such kind of scenarios has been achieved by using pseudo-3D models (parallel 1D components linked by cross-flow junctions) there is room enough for a full 3D system code to improve results.

PARTICIPATION IN THE NEA/OECD MSLB BENCHMARK

Progress in computer technology has made possible to incorporate full 3D modeling of the reactor core into system codes, allowing the simulation of transients in which the interaction between reactor core and plant dynamics is a key factor. The Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA)/Organization for Economic Co-operation and Development (OECD) has released a series of benchmark problems in order to verify the capability of these codes.

The PWR Main Steam Line Break (MSLB) benchmark is a natural continuation of previous NSC Core Transient benchmarks, approved both by the NSC and the Task Group on Thermo-Hydraulic Behaviour, as well as by the Principle Working Group 2 of the Committee on Safety of Nuclear Installations of the NEA/OECD. Its purpose is to validate system best-estimate analysis codes by further verifying their capability to analyze complex transients with coupled neutronics/thermal-hydraulic interactions.

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[1] All of these experiments simulate a main steam line break scenario with different initial containment conditions and with different passive safety systems available

The reference problem chosen for simulation is a hypothetical MSLB in a PWR (Three Mile Island-I) which may occur as a consequence of the rupture of one steam line upstream of the main isolation valves. The asymmetric cooling and an assumed stuck-out control rod during the reactor trip cause significant space-time effects in the core [5].

Due to the hypothetical nature of the transient, the objectives of this benchmark were, on the one hand, to compare the results obtained with different codes by a single user and, on the other hand, to analyze the user's effect by comparing the results obtained with the same code by different users. Some of the codes used by the different institutions taking part were RELAP5-PARCS, RELAP5-QUABOX and RELAP5 3D-NESTLE.

Two different series of calculations performed by UPC members were submitted to the MSLB benchmark, one of them in partnership with the University of Pisa (Italy). QUABOX and PARCS coupled to RELAP5 were used for this purpose [6] [7].

Besides the main conclusion of the analyses performed by the participants, which focused on the safety of the plant in spite of the limitations and anticipated failures imposed to the analysis (rods reactivity value, no boron credit, stuck-out rod, …), other important aspects can be also highlighted. The first one is that best-estimate thermal-hydraulic codes coupled to a neutronics code are able to tackle the simulation of an asymmetrical core cooling in a strongly coupled neutronics-TH scenario. The second one is that the dispersion in the results obtained by the different participants was mainly due to the different thermal-hydraulic modeling of the plant.

RECENT ACTIVITIES RELATED TO 3D SIMULATION

Although some work has been done in relation to 3D simulation for the PANDA experiments, the main efforts in this area have been focused on the Asco-I model.

After the experience acquired in the MSLB benchmark, the UPC developed a RELAP5/PARCS model for Ascó-I power plant [7]. The coupled code RELAP5/PARCS consists of three blocks which communicate through the message passing package PVM: RELAP5, PARCS (developed at Purdue University) and a General Interface that allows the flow of information between the first two blocks.

The thermal-hydraulic part of the model was constructed based on the 1D RELAP5 model of Ascó-I, in which our group has been working for years. As a 1D model of the RPV does not allow the full exploitation of the 3D neutronics code capabilities, a pseudo-3D nodalization (parallel channels) of the vessel was used. Some difficulties appeared when adapting the 1/3-symmetry of the downcomer to the 1/8-symmetry of the core.

The neutronics model is much more detailed: each of the 157 fuel elements plus 9 control rods bancs are simulated. Different neutronic nodes are lumped in a single hydro-dynamic channel, but each fuel element is described by an independent heat structure. So, each single element supplies its simulated power but the thermal-hydraulics feedback is common to a group of fuel elements. Figure 2 shows sketches of the vessel nodalization and of the grouping of neutronic nodes into TH channels.

Limitations arise when trying to validate the thermal-hydraulic component of the model, since many parameters still need to be adjusted. So, it exists also in this subject some room for improvement by using a 3D code.

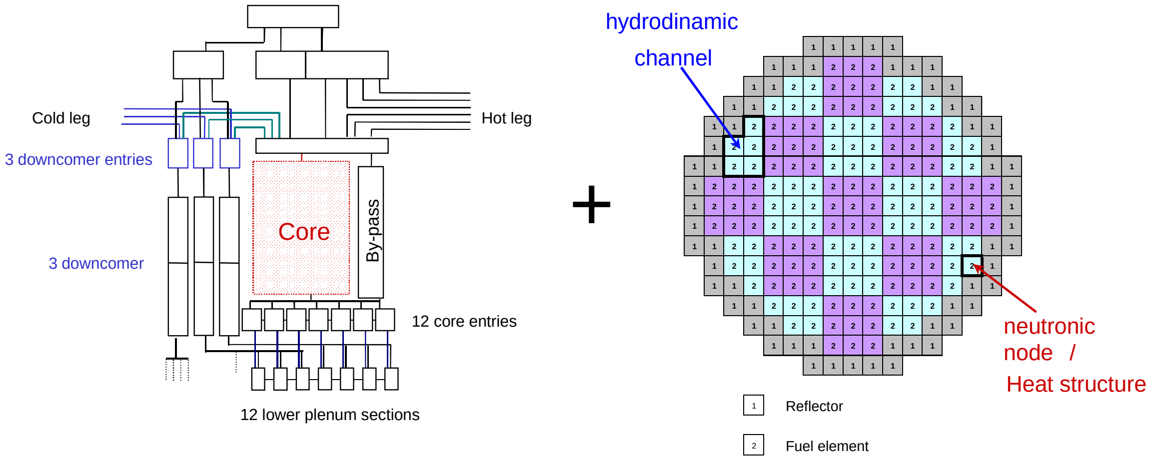

Figure 2: *Pseudo-3D model of Ascó*

FUTURE PERSPECTIVES RELATED TO 3D SIMULATION

As mentioned, the different analyses done during the last years have showed the difficulties when using the available codes to simulate scenarios with important asymmetries. The use of pseudo 3D models helps only partially, as there are too many parameters to adjust (cross flow junctions…). For this reason, our group is interested in having a suitable 3D tool.

Considering the characteristics of the advanced reactor concept, this kind of tool seems to be of major interest in order to simulate the behavior of these plants.

The use of a 3D code may help our group to increase its capabilities for the assessment of safety related issues of commercial nuclear power plants, taking advantage of the experience gained in the past.

Our intention is to perform some validation calculations taking advantage of the experience acquired on commercial power plants. The transient selected for this purpose has been a loss of offsite power occurred at Vandellòs-II, which caused the trip of the reactor, the turbine and the RCPs.

Loss of offsite power at Vandellòs-II

The transient occurred the 24th of august of 1993. The plant was operating at its nominal thermal power (2774 MW at that time) and in nominal conditions when a loss of offsite power caused the trip of all three reactor coolant pumps (RCPs).

Consequently, two seconds later the reactor and the turbine tripped as well. The main feed water system stopped and the auxiliary system began to inject.

Due to the coast down of the pumps, the mass flow in all the loops decreased. The establishment of natural circulation allowed a proper cooling of the core.

Fifteen minutes after the beginning of the transient, and after manually regulating the AFW flow, one of the 3 RCPs was restarted: this produced a flow in loop 3 of 109% of the nominal value and reversed flows in loops 1 and 2. Flow measurements in these last loops were oscillating around 15 % of the nominal value.

Twenty minutes later a second pump was started. This resulted in flow measurements of about 107% in loops 2 and 3, and a reverse oscillating flow of about 30% in loop 1 (figure 3).

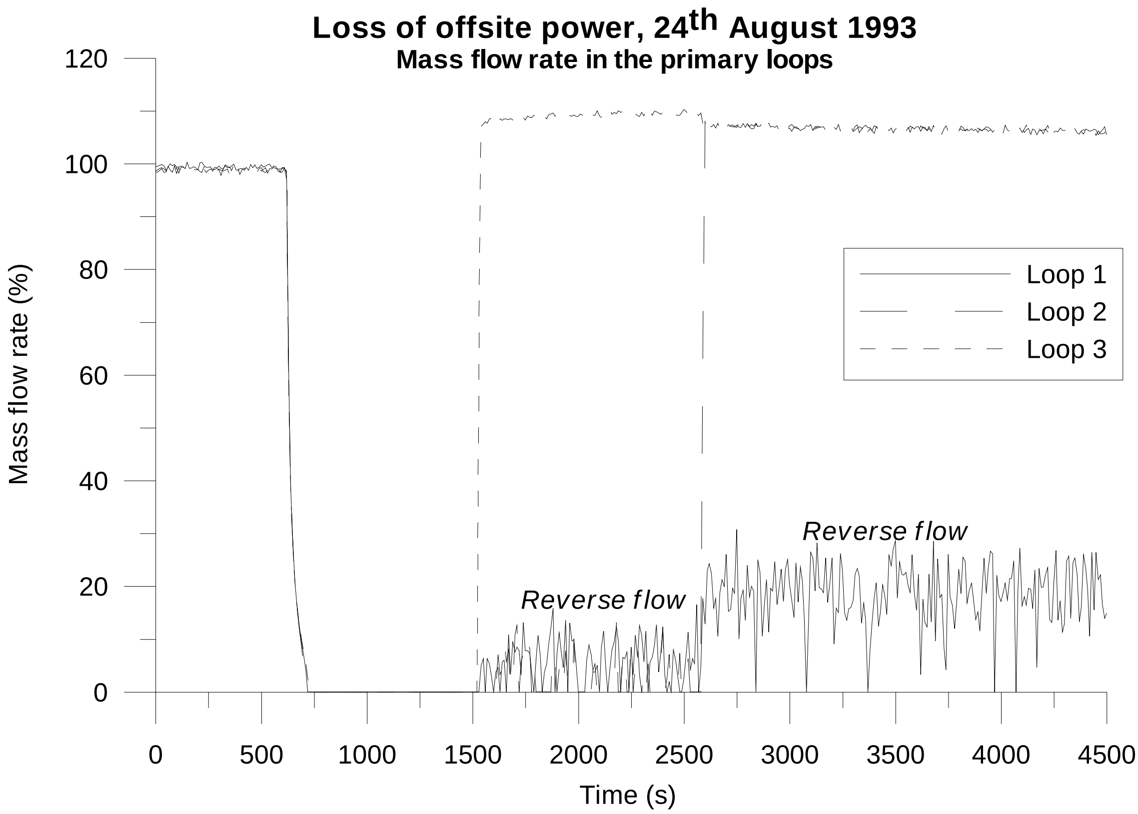

Figure 3: *Mass flow rate during the loss of offsite power transient.*

Objectives of the simulation

The available TH model of Vandellòs-II allows to simulate with high accuracy and precision the behavior of the plant in operational as well as in accidental transients in which the plant ends up in hot zero power.

In the last period, and after the results obtained for Ascó-I, our group has developed a pseudo-3D vessel for Vandellòs-II taking advantage of the similar design and behavior of both plants.

Our purpose is to simulate this fully asymmetrical transient with the RELAP5/Mod3.2 pseudo-3D model of Vandellòs as well as with a new RELAP5-3D model.

The parameters (mass flow, pressure, temperature, delta-T…) recorded at every time step (2 seconds), as well as some steady state temperature map at core outlet, will help to check the correct performance of the model from the thermal-hydraulic standpoint.

The main primary objectives of this simulation are:

- to establish a kind of user’s guide for the adjustment of cross flow parameters in pseudo 3D models
- to assess both plant models by comparing simulation and experimental data
- to evaluate the improvement achieved by using a 3D model instead of a pseudo 3D one.

The final objective of the study is related to safety evaluation. Once the model is assessed, different hypothetical scenarios will be analyzed in order to ensure that design limits are not threatened in loss-of-offsite-power, for an exhaustive set of initial conditions taking into account the number of pumps having restarted as well as more particular aspects like complete or incomplete coast-down and high or reverse flows.

CONCLUSIONS

The UPC has acquired a great deal of experience in the simulation of the three Catalan Nuclear Power Plants and different experimental facilities. This experience is related to operational and accidental scenarios.

An important data base has been compiled at the UPC, after many years of cooperation with utilities and participation in experimental programs.

The results produced in the past by the UPC using the successive released versions of RELAP5, together with the available recorded data, will become very helpful in future tasks of validating and assessing codes and models as well as producing user guidelines.

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