

ANALYSIS OF THE PEACH BOTTOM 2 BWR TURBINE TRIP EXPERIMENT BY RELAP5/3.2 CODE

by

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This paper presents the results of the application of the system of the thermalhydraulic code RELAP5/Mod3.2 in predicting the Peach Bottom Boiling Water Reactor Turbine Trip test. This experiment constitutes a challenge to the capabilities of current computational tools in realistically predicting transient scenarios in nuclear power plants. In fact, it involves strong feedback during the transient between thermalhydraulics and neutronics. In this respect, a reference case was run in order to simulate the interactions between the generated steam line pressure wave propagation and the instantaneous core void distribution. An overall comparison shows good agreement between the code calculations and the experimental data. A series of sensitivity analyses were also performed in order to assess the code prediction features, as well as to identify uncertainties related to the adopted thermalhydraulic parameters used for the plant modelisation.

Key words: BWR turbine trip, Relap5 system code, experimental results, sensitivity analysis

INTRODUCTION

Recent progress in computer technology has increased the possibilities for code calculations in realistically predicting transient scenarios in nuclear power plants. In this context, several attempts have been made in order to enlarge the domain for code application, and to allow best estimate 3D core simulation, including spatial-feedback effects between neutronics and thermalhydraulics. In fact, the incorporation of full-three-dimensional modeling of the reactor core into system codes allows "best-estimate" simulations of interactions between core behavior and plant dynamics, as well as the prediction of dynamic complex transient scenarios envisaged for nuclear power plants. Recently coupled codes were assessed against a hypothetical main-steam line break accident in a PWR [1]. Within

this framework, a BWR computational problem based upon the Peach Bottom Turbine Trip transient tests was selected. The main feature of this case is that it is based on a well-defined experimental test [2, 3]. It also involves strong feedback (self limiting power excursion behavior) during the transient between thermalhydraulics and neutronics. This test is well suited for assessing recently coupled thermal hydraulic systems and 3D neutronic codes for BWR power plant.

The principal aim of this work is to simulate the Peach Bottom 2 Turbine Trip experiment number 2, using coupled codes. In fact, former works shows the feasibility of using the Relap5 code (as a sub-channel code) to represent several channels of the core even in extreme cases as boiloff conditions [4]. The main purpose of the present paper is to test the thermalhydraulic coolant loop system response by fixing the reactor power as an input data vs. time. To do this, a numerical simulation of the experiment was performed. The calculated results were compared with the available experimental ones. The study was then followed up by a series of sensitivity analyses in order to characterize reasons for discrepancies between measured and calculated trends. They also allow the identification of uncertainties related to a hypothesis adopted for the nodalization of the plant.

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TEST FACILITIES AND PROCEDURE

Core configuration

The Peach Bottom 2 is a General Electric designed Boiling Water Reactor. Most of the available reference design data is based on information provided in the two EPRI Reports [2, 3]. The core at the end of cycle 2 was loaded with five kinds of fuel assembly: 576 (7×7 Reload fuel) and 188 (8×8 Lead Test Assembly LTA) fuel assembly type, surrounded by 124 reflector element. The fission chain reaction is mainly controlled by 185 control rods. It is assumed that 2% of the reactor power is released as gamma heating in the in channel coolant

and 1.7% in the core coolant bypass, while the rest is generated in the fuel.

Test description

The experiment consists of manually (by tripping the turbine) closing the Turbine Stop Valve (TSV) at a prescribed operating power level of 61.65% of its nominal value. As a result, a pressure wave is generated in the main steam piping which then propagates with relatively little attenuation into the reactor core. The induced core pressure oscillations result in dramatic changes of the core void distribution and fluid flow. And, due to the inherent negative feedback nature of the core, the

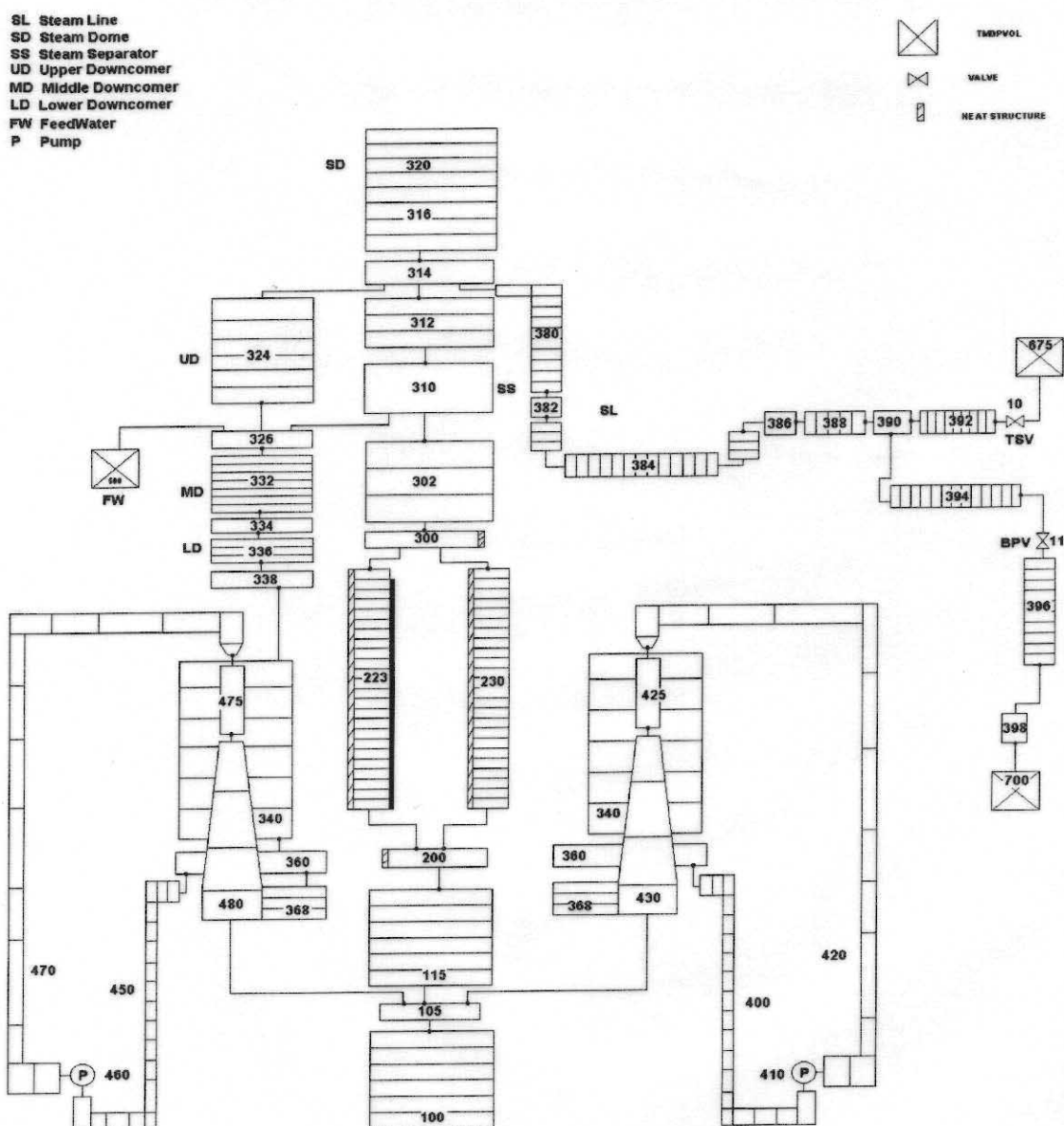


Figure 1. Sketch of the adopted nodalization for Relap5 code

reactor power will rise as a result of void collapsing. A minute upon the second fraction of the TSV closure, the By-Pass Valve (BPV) is opened automatically in order to reduce the pressure rise in the steam line. The turbine stop valve signal that activates the reactor scram initiation was intentionally delayed to allow a relative neutron flux effect to take place in the core. Shortly after the reactor is immediately scrammed.

Main initial and boundary conditions

Basically, the transient begins with the closure of the TSV. The initial reactor power was fixed to of 61.65% of its nominal value (3293 MW). The coolant flow rate was then reduced to 80.9% of its nominal value. The reactor trips when the power level exceeds 95% of its nominal value with a delay time of 0.12 s. To perform the calculations, the reactor power during the transient was imposed; no kinetic calculations were done by the code. It should be noted that during the transient, the resulting pressure rise is not high enough to activate the relief safety valves.

NODALIZATION AND PROBLEM MODELING

In this respect, the core region was modelised using two regions [5]; the active zone, which represents the fuel assembly region, and the core bypass zone. The active zone is thermally represented by heat structures. In the adopted modelisation, the core fuel rods are homogenized into a representative equivalent one. The core was axially subdivided into twenty six (26) meshes; the first and the last mesh correspond to a non-active zone. The remaining 24 meshes represent the active core height. In addition, the four real steam lines were lumped into one, while two jet pumps were used to represent the twenty real ones.

The numerical simulation of the experiment was performed using the RELAP5/Mod3.2 code. The sketch of the adopted nodalization of the Peach Bottom 2 reactor primary coolant system is shown in Fig. 1.

RESULTS AND COMPARISON WITH EXPERIMENTAL DATA

A base case was prepared for the Relap5/Mod3.2 code based upon the Peach Bottom 2 data given in ref. [6] complemented by some experimental data issuing from ref. [3]. First, the code was run for a long steady state period in order to allow most of the calculated transient parameters to reach

stable behavior. At the same time, this step constituted a nodalization qualification test. Next, the transient reference case was run and followed by a comparative study. Several sensitivity cases were carried out in order to identify the dependence of the main parameters during the transient to the adopted initial and boundary conditions.

Steady state

The steady state was run for about 200 s, during which most of the thermohydraulic parameters reached their stable trend. The main achievements are summarized and compared to some available experimental results in Table 1. Some relevant steady state output data, such as the core void fraction axial profile, was well predicted (see Fig. 2). It should be noted that also we have adjusted some singular pressure coefficient losses along the coolant loop in order to reach the same inlet core coolant flow rate as reported in the specifications volume [6]. According to the comparative results shown in Table 1, good agreement between calculations and the experiment was obtained during the steady state, even though a minor calculated core pressure drop was predicted.

Table 1. Main steady state achievement compared with experimental data

Parameter	Calculated	Experimental
Core exit pressure [MPa]	6.7475	6.7647
Vessel dome pressure [MPa]	6.7224	6.7332
Core pressure drop [MPa]	0.0670	0.0836
Inlet core temperature [K]	546.8	545.93
Core average void fraction	0.4065	0.304

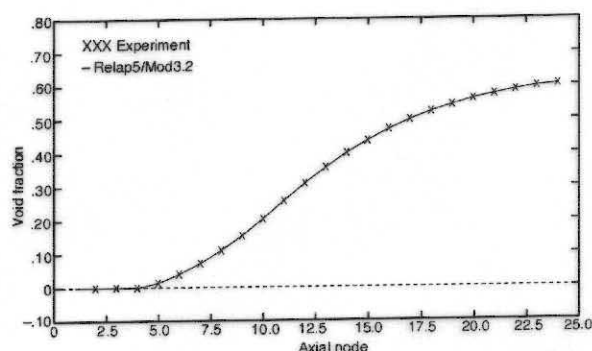


Figure 2. Steady state axial void distribution

Transient state

The transient is quite fast; it lasts for about five seconds. The main sequences take place during the earlier seconds of the transient. The code main results are outlined and confronted to experimental data in Table 2. The calculated and experimental vessel dome pressure evolution during the transient is shown in Fig. 3. The pressure wave in the steam line upstream the turbine is reported in Fig. 4. It can be noticed that this important parameter is well predicted by the code calculation in comparison to the experiment, even though little differences exist in the earlier second fractions of the transient.

Another important parameter is the steam flow rate through the bypass system. It is reported in Fig. 5, compared to the reference flow rate given in ref. [6]. The two evolutions, despite some differences, exhibit the same trend. Differences are essentially due to the way each code models the bypass valve opening; in fact, a linear opening model in time was adopted here.

Table 2 shows the calculated initial pressure response in the dome and in the core exit region, for the whole loop hydraulic inertia during the transient. The calculated initial pressure response is anticipated by about 0.1 s in comparison to the experimental data. The reasons for this discrepancy are not well known, but they seem to be related to the adopted dimension of the plant, which does not exactly reproduce the real coolant flow path length.

Table 2. Main transient achievements compared with experimental data

Parameter	Calculated	Experimental
Maximum vessel dome pressure [MPa (s)]	7.1593 (2.721)	7.1427 (2.98)
Maximum core exit pressure [MPa (s)]	7.1833 (2.735)	7.1839 (2.97)
Maximum steam bypass valve flow [kg/s (s)]	645.86 (0.87)	647.25 (0.87)
Vessel dome pressure initial response [s]	0.28	0.38
Core exit pressure initial response [s]	0.32	0.42
Core exit pressure (after 5 s of transient) [MPa]	7.1409	7.14621
Steam bypass flow (after 5 s of transient) [kg/s]	537.87	507.9

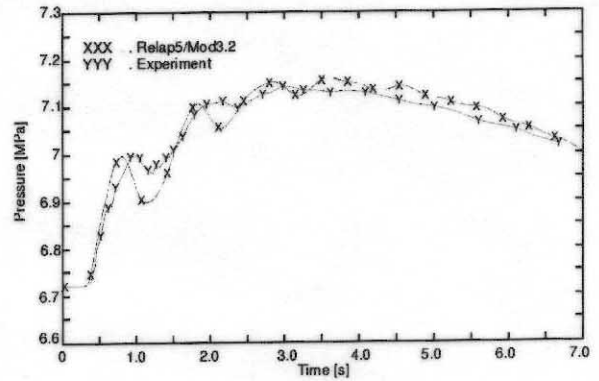


Figure 3. Vessel dome pressure evolution

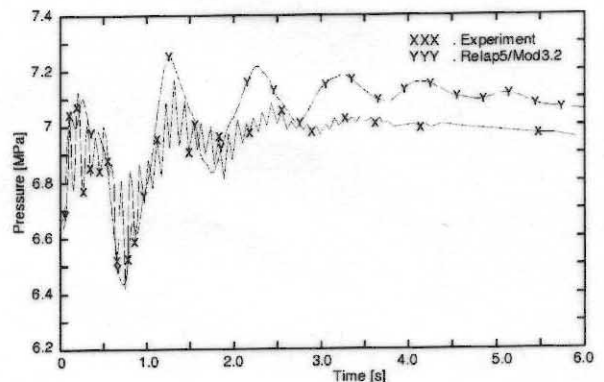


Figure 4. Upstream turbine pressure evolution

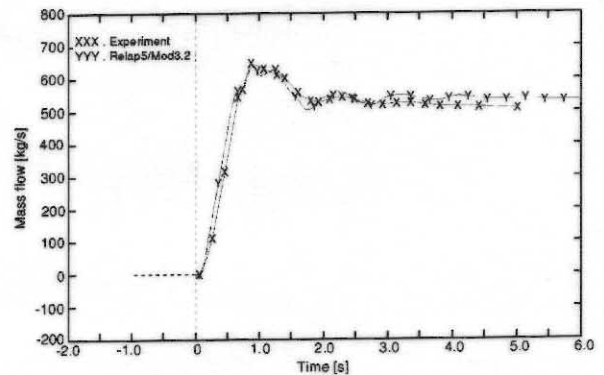


Figure 5. Steam bypass flow rate

SENSITIVITY CALCULATIONS

In order to identify the degree of dependence of the key transient parameters (that govern the transient course) on test conditions, a series of sensitivity analyses have been carried out [7]. The characteristics of the adopted sensitivity cases were chosen in such a way that the steady state remains quasi unchanged. Hence, particular attention was focused on the steam bypass system, which was activated during the transient. The main considered sensitive cases particularly concern uncertainties re-

lated to the vessel steam dome volume, and the steam bypass valve flow area. These two parameters were tuned from 120 to 80% of their reference value, thus 16 cases were performed. The results of such a run are summarized and sketched in Figs. 6 and 7. As can be noticed from Fig. 6, the dependency of the peak core exit pressure has the same asymptotic trend for both parameter changes (steam dome volume and bypass valve area). The variation rate is however more pronounced with respect to the steam bypass flow area change. On the other hand, (see Fig. 7) the peak bypass valve flow rate is found to be proportional to the steam dome change and obviously inversely proportional to the valve flow area. From this sensitivity analysis it is clear that the transient course is closely dependent on the steam bypass system characteristics. In fact, the extreme cases of reducing and enlarging the bypass flow area quasi entirely cover the experimental evolution for both the core pressure and the steam bypass mass flow rate.

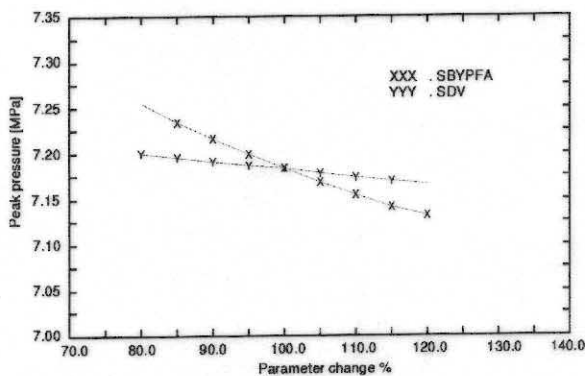


Figure 6. Sensitivity analyses: Effect of steam dome volume (SDV) and steam bypass flow area (SBYPFA) variation on transient peak pressure evolution

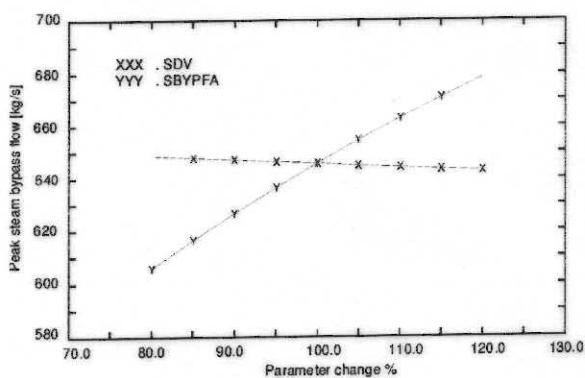


Figure 7. Sensitivity analyses: Effect of steam dome volume (SDV) and steam bypass flow area (SBYPFA) variation on peak pressure steam bypass flow rate evolution

CONCLUSION

The principal aim of the present work is to simulate the Peach Bottom 2 Turbine Trip experiment number 2, using the system code Relap5/mod3.2 coupled with a 3D neutronic code. First of all, it is important to test the whole coolant loop system response by fixing the reactor power as an input data vs. time. This step constitutes the main purpose of the present paper. Detailed results obtained in this respect are presented. The performed calculations are compared against the available experimental data as well as reference data given in the benchmark specification volume. On the whole, the code predicts most of the significant aspects of the transient, such as the pressure wave amplitude across the core, and the steam flow through the bypass valve with acceptable accuracy.

Sensitivity studies have been carried out in order to emphasize the most influential parameters that govern the transient behavior. Several runs showed the degree of dependence of the transient course on the steam bypass system initial conditions. In fact, the experimental data were found to be within the resulting error bands related to extreme cases of reducing and enlarging the bypass valve flow area.

The overall analysis confirms the success of using the Relap5/Mod3.2 system code in simulating the Peach Bottom turbine trip experiment. The core-plant interaction qualification test is almost satisfactory. The next task of this work will consist of simulating the experiment by considering the neutronic-thermohydraulic feedback effects involved during the transient by coupled Relap5 3D neutron kinetics PARCS code.

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Анис БУСБИА-САЛАХ, Франческо Д'АУРИА

**АНАЛИЗА ТРИП ЕКСПЕРИМЕНТА У ТУРБИНИ НУКЛЕАРНЕ
ЕЛЕКТРАНЕ ПИЧ БОТОМ 2 СА BVR РЕАКТОРОМ ОБАВЉЕНА
ПРОГРАМОМ RELAP5/3.2**

У раду су изложени резултати примене термохидрауличких кодова програмског пакета RELAP5/Mod 3.2 у циљу процене трип теста у турбини Пич Ботом електране са BVR реактором. Овај експеримент представља изазов савременим рачунарским средствима у реалистичном предвиђању прелазних сценарија у нуклеарним електранама. Наиме, он укључује јаку повратну спрегу термохидрауличких и неутронских процеса у прелазном режиму. Отуда је разматран референтни случај у циљу симулирања интеракције генерисаног простирања таласа притиска у пароводима и тренутне расподеле шупљина. Целовито поређење показује добро слагање прорачуна и експерименталних резултата. Извршен је, такође, и низ анализа осетљивости у циљу оцене способности програма да предвиди и идентификује неодређености повезане са усвојеним термохидрауличким параметрима коришћеним у моделирању електране.
