NUCLEAR POWER PLANT TRANSIENT SIMULATION USING THERMAL HYDRAULIC / NEUTRON KINETIC COUPLED CODES

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Abstract*. In order to simulate transients that involve core spatial asymmetric phenomena and strong feedback effects between core neutronic and reactor thermal hydraulic, the coupled codes technique is extensively used. The technique consists in incorporating three dimensional (3D) neutron modeling of the reactor core into system codes. In this work, the RELAP5/Mod3.3 thermal hydraulic system code coupled with the PARCS/2.4 3D neutron kinetic code has been used to simulate instability phenomena. In the calculation, PARCS makes use of the moderator temperature and density and of the fuel temperature calculated by RELAP5 to evaluate the appropriate feedback effects in the neutron cross sections. Likewise, RELAP5 takes the space-dependent power calculated in PARCS and solves the heat conduction in the core heat structures. The coupling process between RELAP5 and PARCS codes is done through a parallel virtual machine (PVM) environment. Data from a real BWR nuclear power plant (NPP) have been used as reference conditions and reactor parameters. The coupled codes RELAP5/PARCS have been used to predict the Peach Bottom BWR stability during a recirculation pump trip while the reactor is operating in a special region of power and core flow map. In the simulations, the pump trip did not represent a significant variation in the power evolution and the reactor seems to be very stable in the analyzed cases.*

Keywords*: RELAP5, PARCS, BWR, BWR instability*

1. INTRODUCTION

In the last four decades, the nuclear power industry has been upgrading and developing light water reactor technology, and preparing to meet the future demand for energy. The presently operating Boiling Water Reactors (BWR) contribute with about 21% of the total produced nuclear power worldwide. These plants have reached very ambitious goals of safety and reliability, together with high availability factors, notwithstanding the flow instability and thermal hydraulic oscillations that may affect BWRs under particular operating conditions.

These instabilities can be caused by interdependencies between thermal hydraulic and reactivity feedback parameters such as the void coefficient. BWR transient scenarios, that involve considerable reactivity changes, are described, for example, in the document (OECD, 2004). The document addresses overpressurisation events, large break loss of coolant accidents (LBLOCAs), feedwater temperature decrease, pump trip, increase of core flow, main circulation pump flow rate increase, anticipated transient without scram (ATWS), turbine trip (TT), and control rod removal.

In all BWR transient scenarios, use of coupled 3D techniques is justified by the broad variation in the axial linear power distribution as a function of time. This cannot be predicted by any 0D neutron kinetics model. The recent 3 D nodal neutron kinetic models usually employ planar meshes that are of the size of the fuel assemblies (or part of assemblies). Different coupling code methodologies (Costa, 2007) have been used as, for example, TRAC-BF1/ENTREE, RELAP5-3D, TRAC-BF1/RAMONA, MARS/MASTER, RETRAN-3D, TRAC-BF1/NEM, RELAP5/PANBOX/COBRA, and RELAP5/PARCS.

At the present work, the thermal hydraulic system code RELAP5/MOD3.3 (US NRC, 2001) and the 3D neutron kinetic code PARCS/2.4 3D (Joo *et al.*, 1998) have been used for the simulation of instability transients in the Peach Bottom-2 NPP, while reactor is operating in the region of low-flow/high-power of the power-flow map (Carmichael and Niemi, 1978). The transient herein studied is the recirculation pump trip. The calculated steady state and transient coupled code results are presented and analysed.

1.1. Recirculation Pump Trip Event Description

In the recirculation pump trip event, the stop of a recirculation pump causes a sharp decrease in the core flow, which generates a significant negative reactivity insertion that tends to reduce power and, consequently, the amount of steam generated. This type of event occurred in the LaSalle NPP, in 1988; during a routine surveillance test, an instrument technician inadvertently caused the automatic shut-down of both recirculation pumps. As a consequence, the core flow rate was rapidly reduced from 76% to 29% of rated value, corresponding to natural circulation conditions, and this, in turn, led to the isolation of some of the steam extraction lines leading to the pre-heaters. The result of this action was a colder FW supply to the core. Between four and five minutes after the RPT, the operators observed power oscillations with amplitude range from 25% to 50% of the rated value. The reactor scram occurred automatically on high neutron flux at 118% of rated power at about 7 minutes after the pumps tripped.

2. METHODOLOGY OF ANALYSIS

In this work, the coupled codes RELAP5/PARCS have been used in a coupled way for performing the transient simulation. In particular, the PARCS code is used to evaluate 3D space-time core power history; it uses a non-linear nodal method to solve the two energy group neutron diffusion equations. In the calculation, PARCS makes use of the moderator temperature and density and of the fuel temperature calculated by RELAP5 to evaluate the appropriate feedback effects in the neutron cross sections. Likewise, RELAP5 takes the space-dependent power calculated in PARCS and solves the heat conduction in the core heat structures. The coupling process between RELAP5 and PARCS codes is done through a parallel virtual machine (PVM) environment. A MAPTAB file allows the association among thermal hydraulic and neutronic nodes.

2.1. Three-dimensional Thermal Hydraulic/Neutron Kinetic Model

Peach Bottom Unit 2 is a direct-cycle BWR/4 of General Electric type and capable of generating 1,093 MWe. The radial geometry of the reactor core is shown in the Fig. 1. The core is divided into 15.24 cm wide radial cells, each corresponding to one fuel assembly (FA), plus a radial reflector (shaded area of Fig. 1) having the same width. There are a total of 888 assemblies, being 764 fuel and 124 reflector assemblies. The total active core height is 365.76 cm. The control rods are represented in the Fig. 1 by the crosses. The core control rod bank positions are represented in the figure by seven different colour groups according to the configuration for the steady state conditions. The position represented by ‘‘48’’ represents the bank totally withdrawal.

Peach Bottom was subjected to stability testing. Three turbine trip tests and four series of low-flow stability tests were performed during the first quarter of 1977 at the end of cycle 2. The Peach Bottom nodalization for RELAP5 and PARCS was based on the benchmark specification document for the turbine trip test (Solis *et al*., 2001) and on data in the related tests report (Carmichael and Niemi, 1978). The methodology has been validated against pressure perturbation stability tests (Costa, 2007, Costa *et al*., 2008b) taking use of 33 TH channels. Moreover, others transients as the control rod bank movement event were performed using the coupled nodalisation methodology considered in this work (Costa *et al*., 2008a)

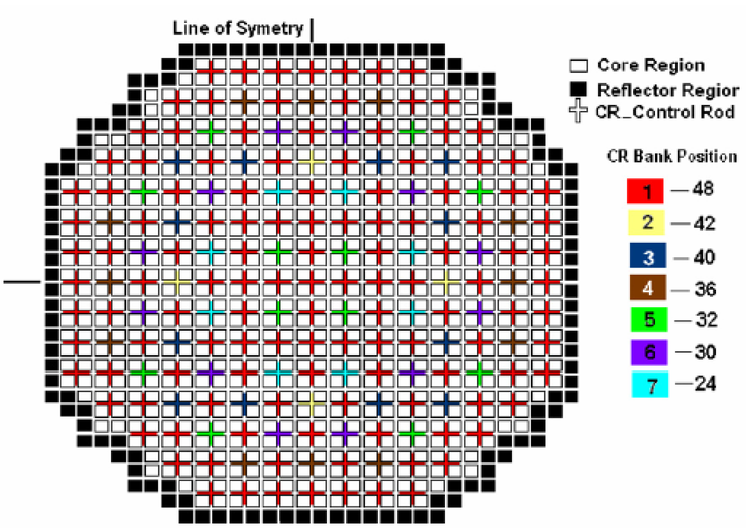

Figure 1. Reactor cross-sectional view.

The Peach Bottom NPP core was divided into 33 heated regions representing the 764 real core fuel assemblies, modelled according to the RELAP5 code requirements; channels with common characteristics were grouped together. In particular, each channel groups a certain number of fuel assemblies; they were chosen according to their thermal hydraulic and kinetic properties, taking into account the lattice type, the relative power, the inlet flow area and the relative position within the core. Figure 2 represents part of the nodalization corresponding to the reactor core; in the figure, the identification number is related to the pipe component in the RELAP5 nodalization. The core active zone was axially subdivided into 24 meshes.

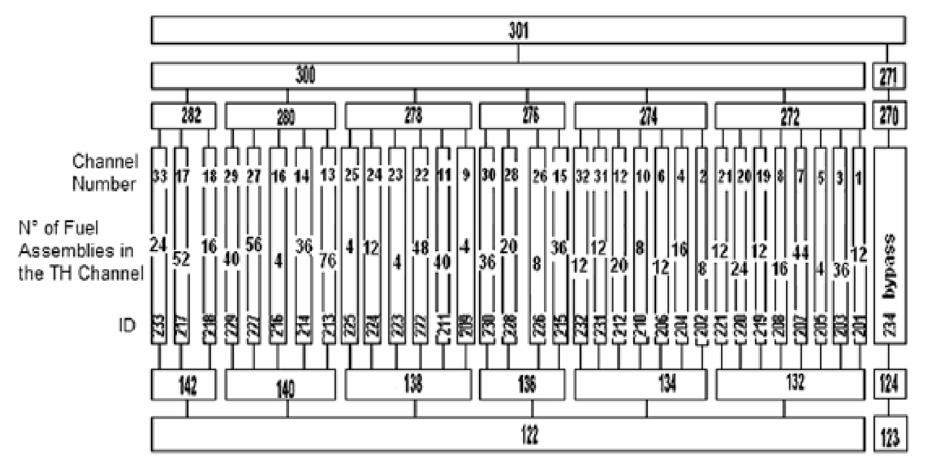

Figure 2. Part of the plant nodalization, with the 33 TH channels in the reactor core.

To represent the reactor core neutronic behavior by the PARCS code, the reactor core was discretized into parallelepipedal nodes, where the nuclear properties are assumed to be constant. Radially, 18 fuel types and one reflector node were defined, whereas axially the core was subdivided into 26 axial nodes; the first and the last nodes represent the reflector zones. In total, 435 compositions or neutronic nodes were considered to represent the kinetic behavior of the core.

In the calculations, steady-state reactor conditions at about 40% core mass flow rate and 59% nominal power (operation point PT3) were assumed as represented in the Fig. 3; that is, operation in the region of the power-flow map was considered, where oscillations have a higher probability to occur. In actual reactor operation, this region is avoided by means of adequately defined control and trip conditions. The core two-phase flow itself provides a potential for oscillatory behavior and the strong feedback between moderator coolant density and core power may enhance this effect under certain conditions.

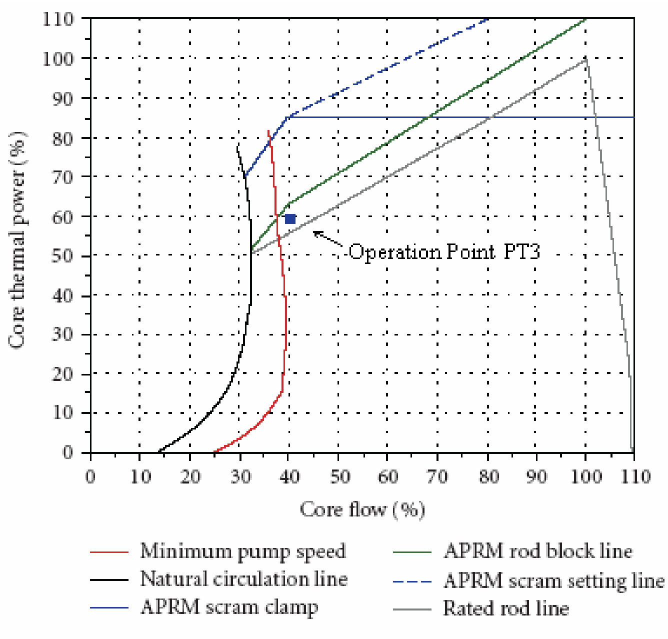

Figure 3. Peach Bottom-2 low-flow stability tests in the power/flow map.

3. STEADY STATE CALCULATION

The steady state simulations were firstly performed using the RELAP5 code stand alone in order to estimate the thermal hydraulic operating conditions under the assumption of fixed and uniform axial power distribution. These initial conditions are then used to perform the coupled calculations. In the coupled steady state calculation, results of the axial power profile were obtained to the operation point PT3 considered in the power/flow map (about 40% core mass flow rate and 59% nominal power) and compared with the available experimental curve with good agreement as it can be verified in Fig. 4.

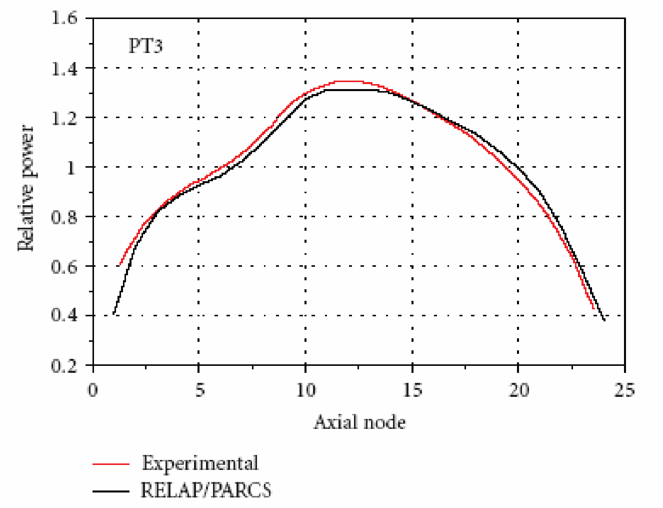

Figure 4. Experimental and calculated mean axial power profile.

4. COUPLED TRANSIENT CALCULATION

There are no experimental data available for comparing the results of the calculations performed for this type of transient, which must be regarded only as a sensitivity analysis. The stop of a recirculation pump causes a sharp decrease in the core flow, which generates a significant negative reactivity insertion that tends to reduce power and, consequently, the amount of steam generated.

To simulate the event, the recirculation pump speed was brought to zero (in the RELAP5 input deck) for one and five seconds, respectively, in two different analyses. In the transient, the pump is shut down for a short time interval and then it is switched on again. The relative power evolutions for the two cases are shown in Fig. 5. One of the two pumps is stopped at the time zero. As it can be seen, the variation in the pump trip duration causes a small variation in the power oscillation amplitude, and the oscillations are terminated at the same time.

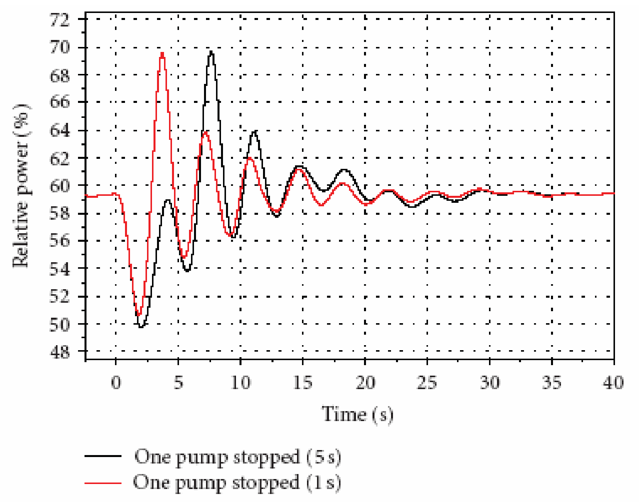

Figure 5. Relative power evolution considering one recirculation pump stopped for two different time intervals: 5 seconds and 1 second.

In addition, another case was considered in which both pumps were stopped, at the same time, for one second. As it can be noted in Fig. 6, the amplitude of the power oscillation in this case is higher, as it is expected to occur. The periods of oscillation, for both cases, are practically the same; the reactor reaches stability nearly in the same time for these two cases.

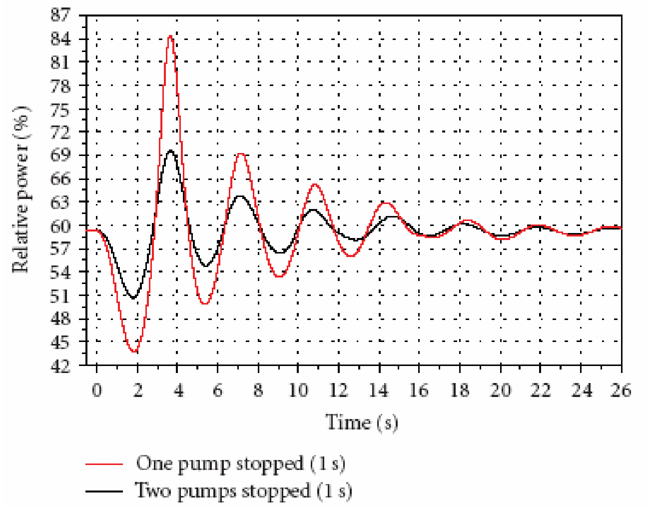

Figure 6. Relative power evolution after recirculation pump trip.

The core power exhibits damped in-phase oscillations with a decay ratio value less than 1.0, characterizing a stable system after the transient event. The results of DR (decay ratio) and NF (natural frequency) to the power oscillations are presented in Tab. 1. DR and NF were calculated by a specific algorithm and presents values very similar to those found for the case of the pressure perturbation (Costa *et al.*, 2008b). The decay ratio and the natural frequency of the oscillations are typical parameters used to evaluate the instabilities. Other parameters can also provide valuable information, such as the Lyapunov exponents associated to the time series. In fact, Lyapunov exponents are also used as a measure of the stability of the neutronic time series (Pereira *et al.*, 1992).

Table 1. DR and NF for power oscillation in the pump trip event.

Calculation case	DR	NF (Hz)
(1) One pump stopped for 1 s	0.427	0.278
(2) Two pumps stopped for 1 s	0.345	0.280

As it is shown in the Fig. 6, after perturbation, reactor power has a fast decrease in its value because of the negative reactivity inserted in the core. The sharp decrease of the core flow (Fig. 7) causes a high negative reactivity insertion into the core due to void fraction increase (Fig. 8). After 1 s, the pumps' velocities return to initial values and, due to void fraction decreasing and consequent positive reactivity insertion, power rises up to 84.3 %. Then, more steam is produced and the value of the void fraction rises again; consequently, there is a negative reactivity insertion, causing power to decrease. This process presents a fast decrease in amplitude oscillation and, after approximately 20 seconds, oscillations are terminated and reactor returns to the initial power level.

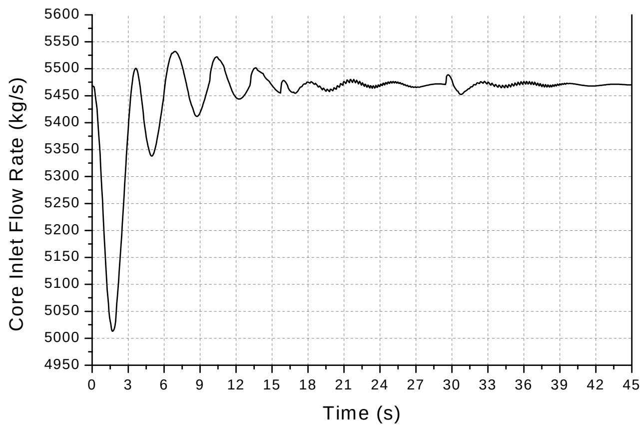

Figure 7. Core inlet mass flow rate evolution.

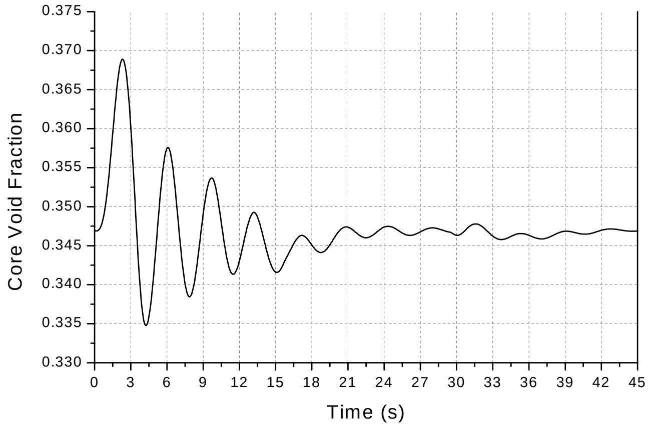

Figure 8. Core void fraction evolution at mid height (axial level 12).

4. CONCLUSIONS

In this work, RELAP5/MOD3.3 thermal hydraulic system code and the PARCS-2.4 3D neutron kinetic code were coupled to simulate pump trip transients in a BWR. The coupled system was firstly validated for the test point PT3 in steady-state conditions. The pump trip event has been considered as only a sensitivity analysis and was simulated using the same operating conditions of the PT3 experimental operation point.

In the simulations, the pump trip did not represent a significant variation in the power evolution and the reactor seems to be very stable in the analyzed cases. The core power exhibited damped in-phase oscillations with a decay ratio value less than 1.0, characterizing a stable system after the transient event.

5. ACKNOWLEDGEMENTS

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