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VALUATION OF CHANGES IN THE CLADDING AND FUEL TEMPERATURES AFTER AN EXTREME TRANSIENT CASE OF CONTROL ROD BANKS WITHDRAWAL IN A BWR

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Abstract. *Instabilities in BWR (boiling water reactor) are possible to occur when an operating condition becomes unstable after some change in system parameters. As a consequence, state variables identifying the reactor working conditions are observed to oscillate in different ways depending on the modalities of the departure from the stable operating point. In this work, the RELAP5/MOD3.3 thermal-hydraulic system code and the PARCS/2.4 3D neutron kinetic code were adopted to simulate coupled instability phenomena in the Peach Bottom BWR. In the transient investigated, the control rod banks (CRB) are continuously removed from the core starting from 20 s (steady state) up to 40 s. The reactor was brought to unstable behaviour. The power remained with a behavior approximately constant up to about 90 s but it begins to oscillate with amplitudes reaching more than 80% of total power. The cladding temperature increases drastically in one extreme of a selected fuel assembly (axial level 3) after the rod banks are removed. This phenomenon is directly connected with the change in axial power distribution, which is drastically affected by the rod banks withdrawal. Since after rod withdrawal, the coolant density is much higher at the bottom core inlet, the expected bottom-peaked power profile is observed. The fuel temperature rises drastically, at the level 3. Temperatures of about 1200 K were observed. These values are below the melting point of the fuel (≈ 3073 K) during this transient. In the calculation, the scram intervention was not considered, because the main interest was to assess the core parameters evolution during an extreme event.*

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

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 (NEA, 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.

Power oscillations can, for large amplitudes, have an unwanted influence on the fuel integrity. In the fuel temperature limitation, it is essential to prevent the exceeding of the melting point (3073.15 K for UO_2). Fuel elements subjected to temperatures sufficiently high to induce centreline melting will experience a significantly higher probability of failure (loss in the functional behavior caused by a change in the physical properties). Furthermore, the low thermal conductivity of ceramic fuels leads to high temperature gradients that can cause fuel cracking and swelling (Duderstadt and Hamilton, 1976).

In all BWR transient scenarios, the application 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 have been used, for example, TRAC-BF1/ENTREE, RELAP5-3D, TRAC-BF1/RAMONA, MARS/MASTER, RETRAN-3D, TRAC-BF1/NEM, RELAP5/PANBOX/COBRA, RELAP5/PARCS, and several others.

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, a BWR type, while the reactor is operating in the region of low-flow/high-power of the power-flow map (Carmichael and Niemi, 1978). The transient addressed in this work was simulated by the withdrawal of control rod banks from the core. The reactor was brought to unstable behavior before the end of the partial control rod banks withdrawal. The power oscillations remained until the end of the calculation. The thermal hydraulic (TH) channels number 27, 60, 93 and 126 were taken as reference to demonstrate the behavior of some TH parameters in different quarters of the core during the transient. In the nodalization, these channels are symmetrically localized in the core with respect to the core centre. The trend of the mass flow rate, pressure, coolant temperature and the void fraction for the four thermal-hydraulic channels were taken. The behavior of the TH parameters investigated presented out-of-phase evolution comparing the channels that represents each half of the core. The channels number 27, 60, 93 and 126 have been selected to investigate the thermal hydraulic core behavior because each one is composed by 14 fuel assemblies and the transient effect evolution in such channels becomes more visible. The channels 5, 38, 71 and 104, also localized in different quarters of the core, were chosen to valuate the evolution of the temperatures in the fuel and in the cladding because they are connected with only one fuel assembly.

2. THREE-DIMENSIONAL THERMAL-HYDRAULIC/NEUTRON KINETIC MODEL

The Peach Bottom nodalisation for the RELAP5 and PARCS code was developed at the University of Pisa and it 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 was validated in relation with the four pressure perturbation stability tests (Costa *et al.*, 2008) for a model with 33 TH channels.

In the new model, the core was divided into 132 heated regions representing the 764 real core fuel assemblies, modeled according to the RELAP5 code requirements. The planar TH channels distribution is shown in the Fig. 1. The channels highlighted in the Fig. 1 in gray color (27, 60, 93 and 126) have been selected to investigate the thermal hydraulic core behavior at each quadrant during the transient events. These channels were selected to the analyses because they are composed by 14 fuel assemblies and the transient effect evolution in such channels is more visible. The channels highlighted in blue (5, 38, 71 and 104) were chosen to valuate the evolution of the temperatures in the fuel and cladding in only one fuel assembly.

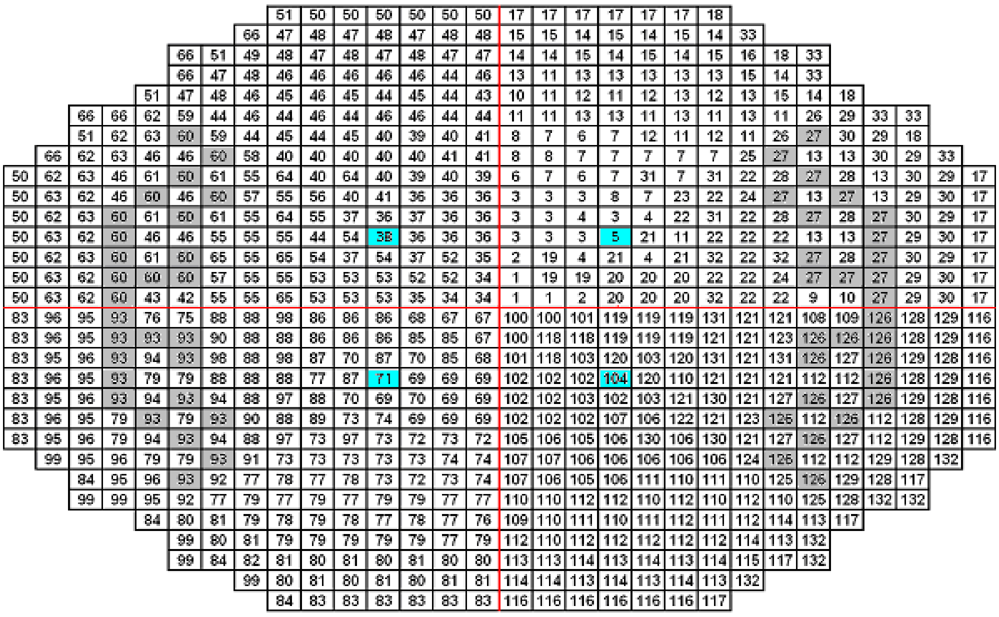

Figure 1. The 132 TH channels in the reactor core modeled for the RELAP5.

The core active zone was axially subdivided into 24 meshes. To represent the reactor core neutronic behavior by the PARCS code, the 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.

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.

3. RESULTS FROM STEADY STATE CALCULATION

In the calculations, steady state reactor conditions at about 40.0% core mass flow rate and 59.0% nominal power were assumed, characterizing operation in the region of the power-flow map where oscillations have a higher probability. In the Fig. 2, the result of the coupled calculation for the mean axial power profile for the 132 core channels configuration is sketched in comparison with the measured one, presented in (Carmichael and Niemi, 1978), and also with the calculated one for 33 core channels configuration (Costa *et al.*, 2008). As it can be observed, the measured and calculated axial mean core powers are in good agreement, in spite of the new configuration (132 channels) slightly overestimated the axial power in the central region of the core.

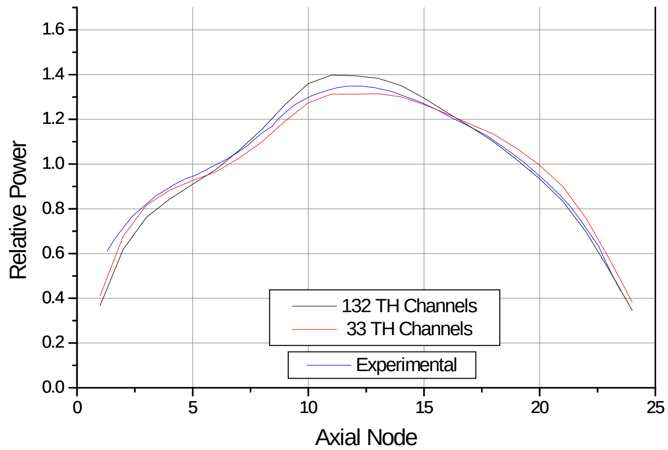

Figure 2. Measured and calculated mean axial relative power profile.

The Fig. 3 illustrates the 2D relative power distribution at the axial plane 12. As the figure shows, the power peak is located in a peripheral zone. This radial flux asymmetry makes the use of coupled code calculations more valuable and necessary for a reasonable phenomena prediction. The power distribution is essential for the subsequent thermal-hydraulic analysis of the core, since the core behavior as well as its fuel depletion will affect both core composition and microscopic cross sections.

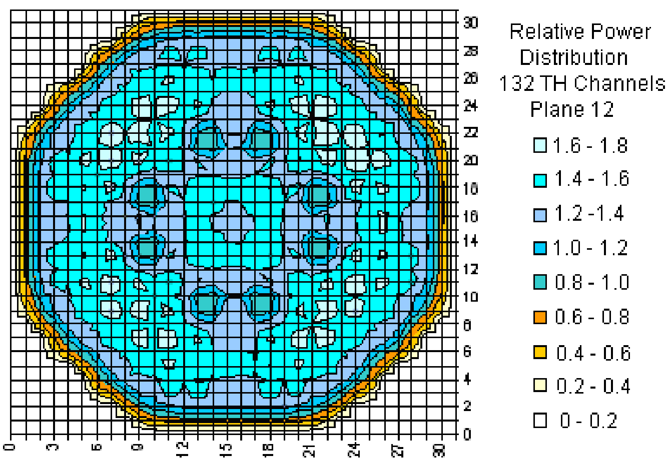

Figure 3. Radial 2D core power distribution (axial level 12) for the configuration with 132 TH channels.

4. RESULTS FROM TRANSIENT CALCULATION

The core presents a total of 888 assemblies, being 764 fuel and 124 reflector assemblies (represented in the Fig. 4). The total active core height is 365.76 cm. The control rods are represented in the Fig. 4 by the crosses and their positions are represented in the figure by seven different color groups according to the configuration for the steady state conditions. The position represented by ‘‘48’’ represents the bank totally withdrawal.

The transient addressed was simulated by the withdrawal of some selected control rod banks from the core. Firstly, the coupled calculation was performed to achieve stable steady state conditions, with the seven rod banks in the configuration shown in the Fig. 4 and also in the Tab. 1 (t = 0).

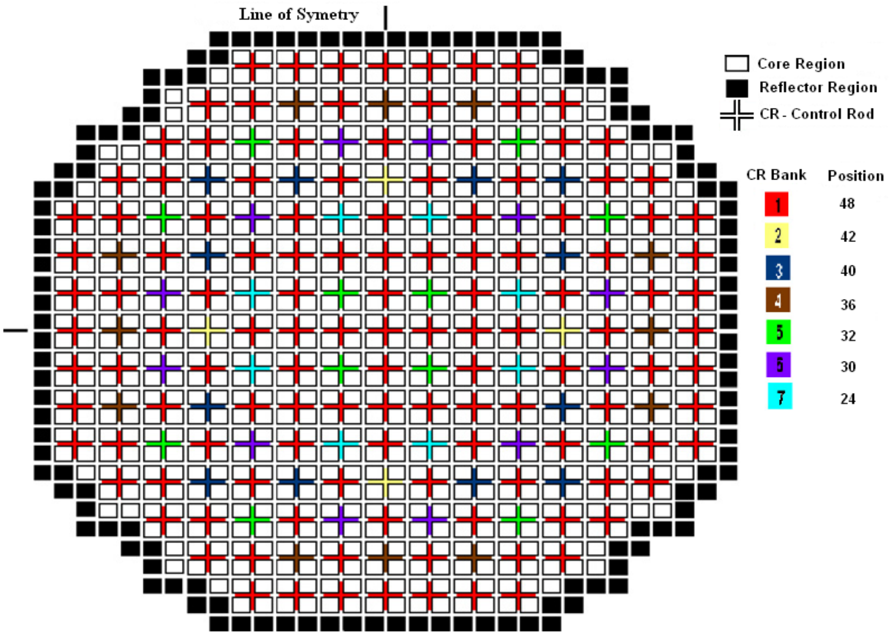

Figure 4. Reactor cross-sectional view with the CRB distribution and their steady state positions.

In the coupled transient calculation, the control rod banks were continuously removed according with the following way (also described in the Tab. 1): starting from 20 s of calculation, all the control rod banks were 6 positions withdrawal from 20 up to 40 s. From 40 up to 300 s, the rod banks remained with the same configuration that they had in the time 40 s. Sections 4.1, 4.2 and 4.3 present the effects in the power, TH related parameters and fuel temperature, respectively, in consequence of such perturbation event.

Table 1. Sequence of withdrawal of the control rod banks.

	Steady State		Transient	
	t = 0 s	t = 20 s	t = 40 s	t = 300 s
	CRB Position		CRB Position	
Bank 1	48	48	48	48
Bank 2	42	42	48	48
Bank 3	40	40	46	46
Bank 4	36	36	42	42
Bank 5	32	32	38	38
Bank 6	30	30	36	36
Bank 7	24	24	30	30

4.1. Power evolution

Figure 5 shows the power evolution for the investigated case. As it can be verified, there is a small power perturbation when the control rod banks were 6 positions withdrawal during the short period of 20 s. The withdrawal stops at 40 s of calculation and the power tends to stabilize. The power remains approximately constant up to about 90 s but it begins to oscillate with amplitudes reaching more than 80% of total power. After 150 s, the oscillations became stronger with amplitude levels of about 120% up to the end of the calculation. It is necessary a more detailed study to try to find the possible causes of the power instability. The related thermal-hydraulic parameters (mass flow rate, pressure, void fraction and coolant temperature) have been investigated and their behaviors are presented next.

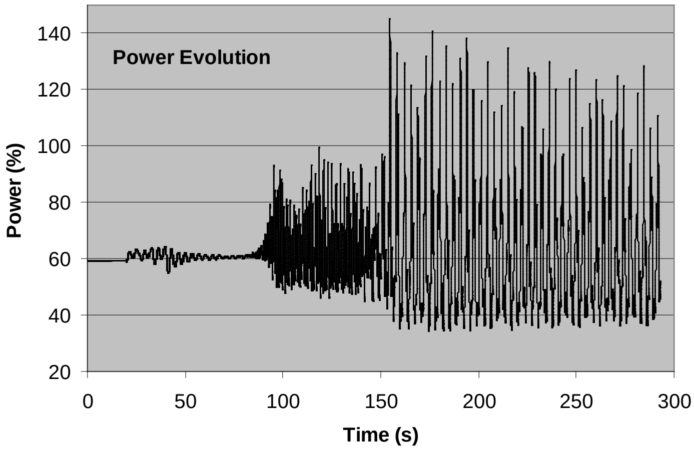

Figure 5. Total power evolution for the addressed case.

4.2. Thermal hydraulic parameters evolution

Four TH core channels, 27, 60, 93 and 126, were taken as reference to demonstrate the behavior of the thermal hydraulic parameters during the transient. In the nodalisation, these channels are localized in four different quadrants of the core and they are symmetric with respect to the core centre. The Fig. 6 presents the trend of mass flow rate to the four TH channels. Observing the parameter evolution in greater detail (Fig. 7) it is possible to verify an out-of-phase behavior: when the mass flow reaches a maximum value in a half of the core (channels 27 and 126), in the other it presents a minimum value (channels 60 and 97). The analysis demonstrates an oscillatory behavior with a frequency of about 0.5 Hz, typical of this type of transient. As can be observed the channels 60 and 97 are not exactly in phase. Further investigations will be performed to find the causes of this behavior.

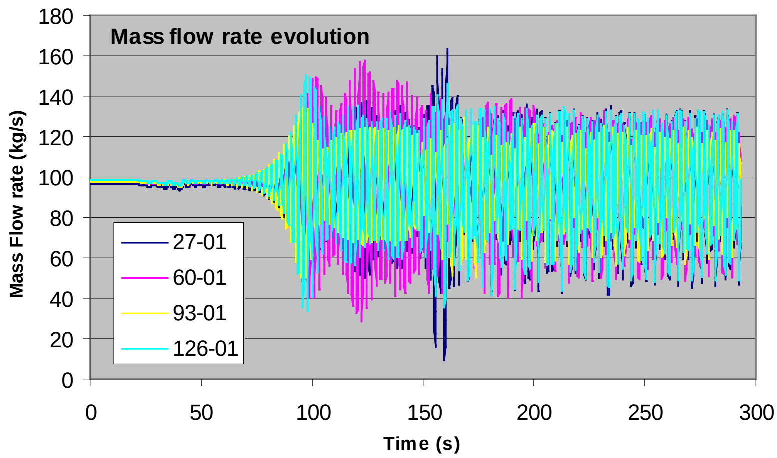

Figure 6. Inlet mass flow rate evolution to the four selected TH core channels localized in different quarters of the core.

The inlet pressure evolution (at the axial level 1) to the four channels is presented in the Fig. 8. As it can be observed, the pressure oscillates in-phase for all the analyzed channels. That is, the pressure has a constant value in all points of an axial plane of the core for a given calculation time. The pressure drop across the core was observed to be approximately constant in time and space (considering that each channel represents different quarters of the core). The fact that the large number of parallel channels of a BWR has common lower and upper plena imposes the same pressure drop to all the reactor fuel channel assemblies (March-Leuba and Blakeman, 2001).

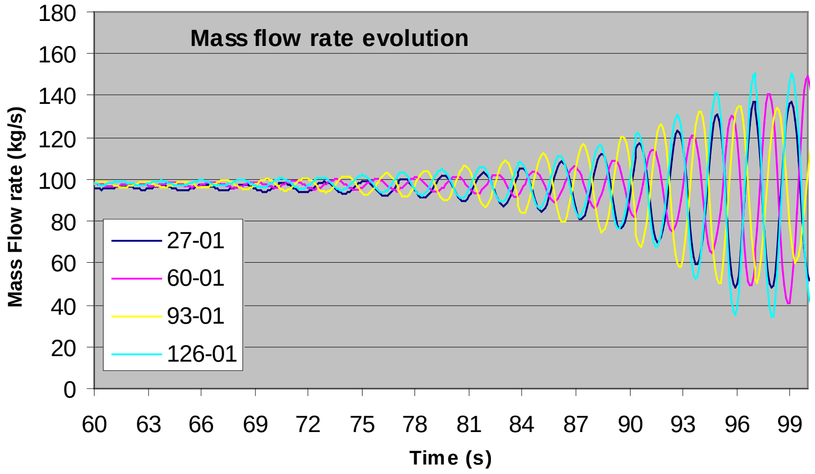

Figure 7. Inlet mass flow rate evolution to the four selected TH core channels from 60 up to 100 s time window.

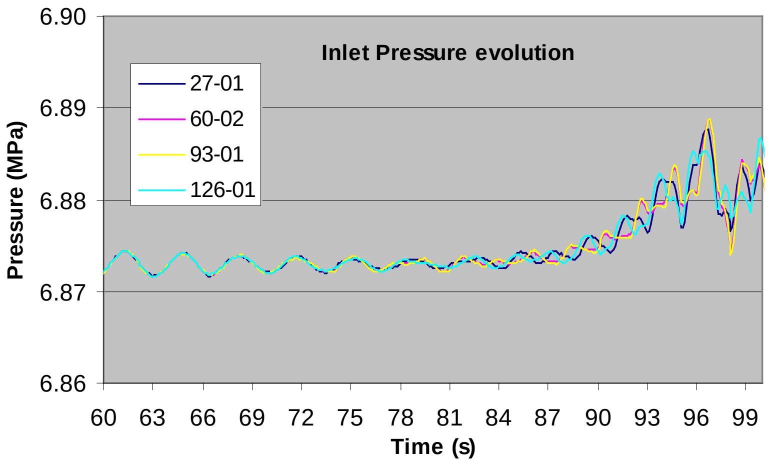

Figure 8. Inlet pressure evolution from 60 up to 100 s time window.

In addition, the Fig. 9 represents the void fraction evolution in the time window at the mid height to the four channels. Also this parameter is found to behave out-of-phase if considering two halves of the core. As it occurred with the mass flow rate, also the void fraction evolutions in two quarters of the core are approximately in phase (channels 60 and 93) but they are out-of-phase in comparison with the other half of the core represented by the channels 27 and 126. The void fraction begins to oscillate at approximately the same time in that it occurred with the mass flow rate. The same behavior is observed for the coolant temperature evolution (Fig. 10).

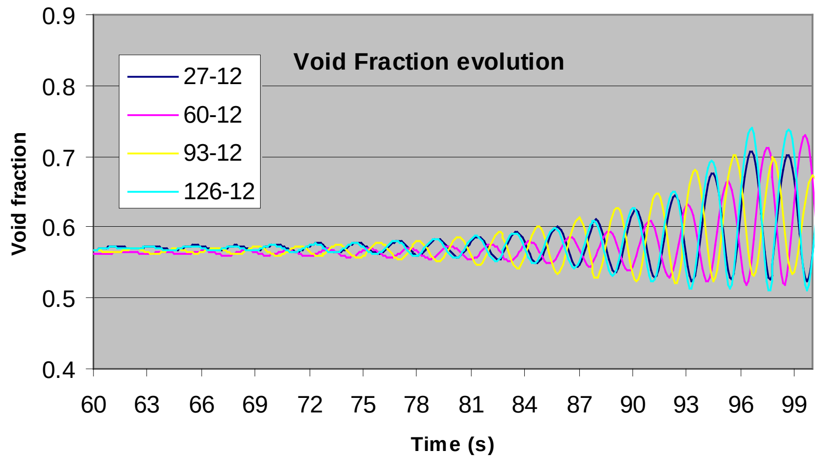

Figure 9. Void fraction evolution at mid height (axial level 12) from 60 up to 100 s time window.

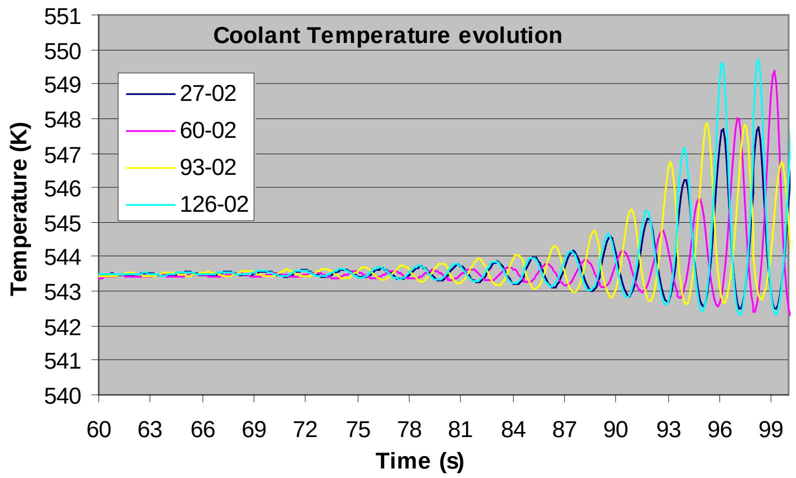

Figure 10. Inlet coolant temperature evolution from 60 up to 100 s time window.

4.3. Fuel and cladding temperature analysis

Figure 11 shows the fuel centreline temperature evolution in the channel 05 at eight different axial levels. The fuel temperature rises drastically, at the level 3, as a consequence of the rod bank withdrawal from 20 to 40 seconds. Temperatures of about 1200 K were observed. These values are below the melting point of the fuel (3073 K) during this transient. Obviously, in the calculation, the scram intervention was not considered, because the main interest was to assess the core parameters evolution during an extreme event.

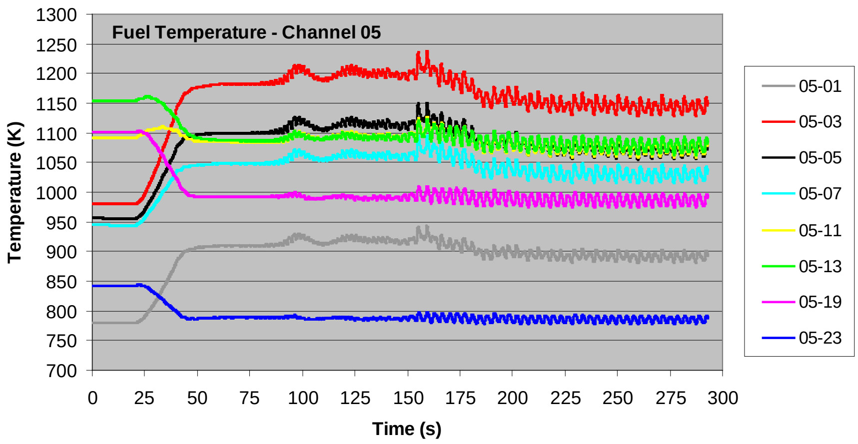

Figure 11. Centreline fuel temperature evolution in the channel 05 at eight axial levels.

Figure 12 shows the cladding temperature evolution at several axial levels for the TH channel number 5. As it can be observed by the Fig.12, the cladding temperature tends to increase in central part of the fuel assembly (axial levels 3, 5, 7 and 11) after the rod banks are partially removed. However, the change in the cladding temperature along the assembly is small (from a minimum average value of about 563 K up to a maximum of about 569 K). Moreover, in spite of the fuel temperature to vary drastically with the CRB withdrawal, the cladding temperature is not affected in the same way remaining practically with the same values of the steady state temperature.

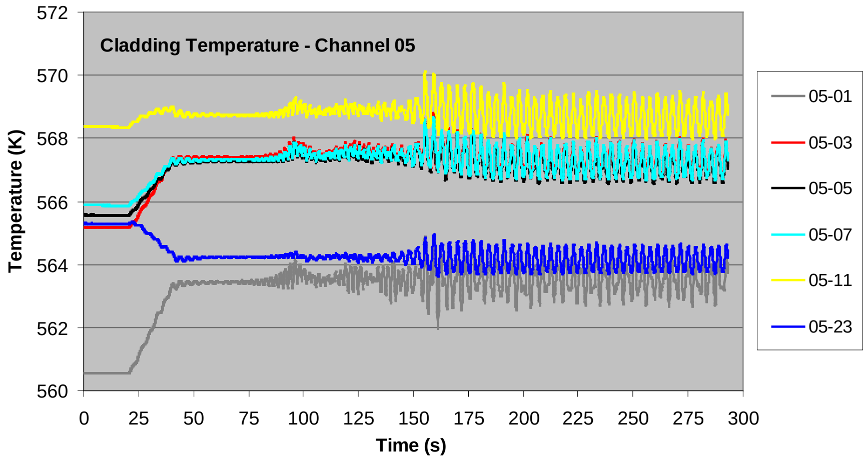

Figure 12. Cladding temperature evolution in the channel 05 at six axial levels.

In the Fig. 13 it is possible to verify the evolution of the fuel centreline temperature in the channels 5, 38, 71 and 104 at the axial level 3. The respective curves present, approximately, the same behavior. Figure 14 shows the average axial fuel temperature profile for the channel 5 in steady state and transient conditions. During the transient, the fuel temperature increases drastically in one extreme of the heat structure (axial level 3) after the rod banks are removed, as can be observed in the Fig. 14. This phenomenon is directly connected with the change in axial power distribution, which is drastically affected by the rod banks withdrawal as it can be seen in Fig.15. Since after rod withdrawal, the coolant density is much higher at the bottom core inlet, the expected bottom-peaked power profile is verified.

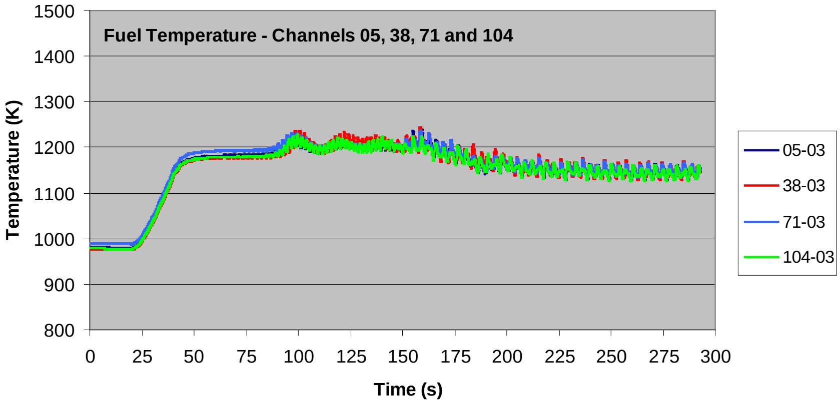

Figure 13. Centreline fuel temperature evolution in the channels 5, 38, 71 and 104 at the axial level 3.

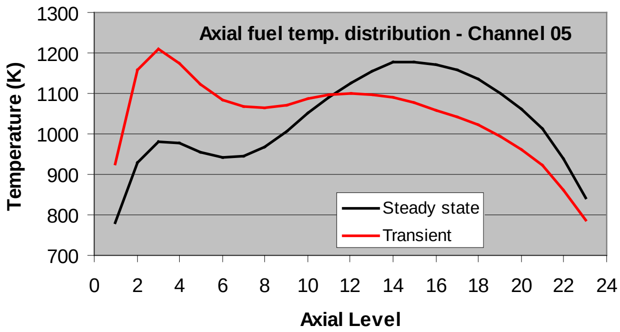

Figure 14. Axial fuel temperature in the channel 5, for steady state and transient cases.

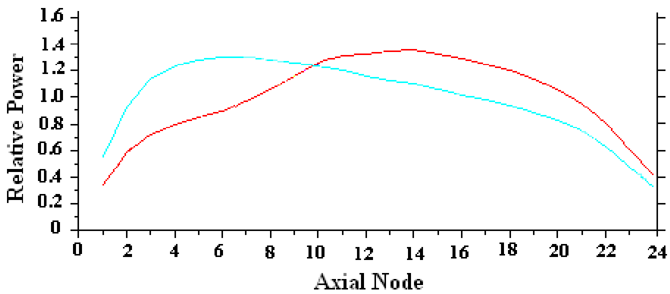

Figure 15. Axial relative power distribution for steady state (in red) and for transient conditions (in blue).

5. CONCLUSION

In this work, RELAP5 and the PARCS codes have been adopted to simulate coupled instability phenomena occurring as a consequence of CRB withdrawal in a BWR. An hypothetical case was investigated and the event was considered only as a sensitivity analysis. In the steady state analysis, the measured and calculated axial mean core powers presented very good agreement.

In the transient coupled simulation, the CRB were continuously removed starting from 20 s of calculation; all the CRB were 6 positions withdrawal from 20 up to 40 s. From 40 up to 300 s, the rod banks remained with the same configuration that they had in the time 40 s. The fast removal of the CRB by 6 positions, brought the reactor to instability. Small amplitude oscillations were observed in the power evolution during de rod banks withdrawal. After the transient (at 40 s), the power presented a stabilized behaviour but it begins to oscillate again with amplitudes reaching more than 80% of total power. After 150 s, the oscillations became stronger with amplitude levels of about 120% up to the end of the calculation in a limit cycle.

The investigations of the related TH parameters showed that the mass flow rate, the void fraction and also the coolant temperature begin to oscillate at approximately the same time interval. A more detailed study will be necessary to find the actual causes that to trigger of the power oscillations. A possibility is to investigate the local power evolutions at different core points and levels to have a more precise idea respect to the local power behavior.

Fuel temperature of about 1200 K was verified after the transient. However this value is below the melting point and then the event does not represent risk to induce fuel failure for this especific perturbation simulated.

6. ACKNOWLEDGEMENTS

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