TWO-PHASE FLOW INSTABILITIES ANALYSES IN BOILING WATER REACTOR

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Abstract. *The problem of two-phase flow instabilities is frequent in several engineering systems such as the Boiling Water Reactors (BWR). BWR instabilities occur when an operating condition becomes unstable after some change in system parameters. As a consequence, state variables identifying the reactor working conditions (as the system temperature, pressure and mass flow) are observed to oscillate in different ways depending on the steady state condition of the reactor affecting the power behavior. Out-of-phase oscillations are an important type of instability and its study is relevant for its potential safety implications; in fact, during out-of-phase instabilities, local power oscillates in a self-sustained way so that average reactor power remains nearly constant. In this work, the RELAP5 thermal-hydraulic system code coupled with the PARCS neutron kinetic code have been used to simulate the instability phenomenon. Data from a real BWR nuclear power plant (NPP) have been used as reference conditions and reactor parameters. Typical parameters used to evaluate the instabilities such as the Decay Ratio (DR) and the Natural Frequency (NF) have been analysed and presented in this work.*

Keywords*: Stability BWR, Decay ratio, Natural Frequency*

1. INTRODUCTION

The stability of a reactor using a biphasic fluid as a refrigerant, in this case, a boiling water reactor (BWR), has always been susceptible to change of stability. Stability is understood as the ability of a reactor return to stationary state when there has been a small perturbation in a parameter related to the dynamics.

To prevent instability events of the power, it could monitor the Decay Ratio (DR). The stability of a boiling water reactor is usually characterized by the DR parameter, which quantifies the degree of damping or amplification which occurs in the system when a given disturbance is experienced, which may be due to 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 this paper, the thermal-hydraulic system code RELAP5/MOD3.3 (US NRC, 2001) and the 3D neutron kinetic code PARCS/2.4 (Barber, 1998) have been used for the simulation of instability transients, which consists of the abrupt increase of pressure in the turbine of the reactor. The calculated steady state and transient coupled code results are presented and analyzed.

The operating conditions of a BWR core are commonly represented in the flow-power map that correlates the percentage of thermal power of the core with the percentage of flow in the core. The minimum flow is determined by both natural circulation conditions as the minimum speed of the recirculation pumps.

Nominal operating procedures, for controlling the power percent are essentially based on flow changes by varying the pump speed, maintaining constant control rod configuration. Although strategies for optimizing operation of the core, fast power changes, and load tracking form, takes into account the setting motion of control rods within the power control.

Bulletin 88-07 of the US regulatory agency (NRC) "interim corrective actions" (US NRC, 1988), which defined by a map of "flow-power" three regions of operation in order to restrict operating, the operator place in the area of greatest risk, to avoid introduced to one of these areas, develop immediate actions, as appropriate. In the Fig. 1 is shown the regions where the main actions to be taken are: Region C: is permitted only during the startup process reactor; Region A and B: it requires immediate action to get out of these areas, excluding restarting the recirculation pumps.

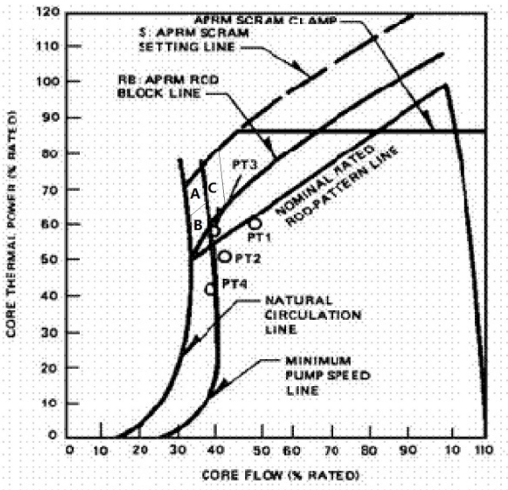

Figure 1: Peach Bottom-2 low-flow test conditions: three specific regions A, B and C.

In the experiment, the magnitude of the pressure set point steps was selected at approximately 8 psi (0.22 MPa) which gave a good signal-to-noise ratio in the neutron flux response and did not cause operational difficulties during the testing. Then, it is considered a small pressure perturbation at point 3. The main parameters of such test are shown in the Tab. 1.

Table 1. Experimental conditions point 3 – PT3 in the Fig. 1.

Power		**Mas flow rate (core inlet)**		**Enthalpy (core inlet)**	**Pressure (core inlet)**
%	MW	kg/s	%	kJ/kg	Mpa
59.2	1948	5216.4	40.4	1184.6	7.10

3. QUANTIFICATION OF INSTABILITY: DECAY RATIO

Have been developed many investigations concerning to BWR instability, because these are likely to fall into regions of instability. In order to better understand the stability of these reactors have been developed three forms of analysis. The first of these is related to the attempt to identify the reasons for occurrences of instability in terms of basic physical, knowing the variables physical, the couplings and the mechanisms between the various system parameters. This approach uses several simplifications (from the point of view of the system). The second way is conceptually related to the first; in this case, attempting to calculate the parameter known as decay ratio (DR) from large core physics - thermal hydraulic system codes, (Analytis, 2001; Hotta 1997), the decay ratio can be successfully calculated using such codes and its correctness was seen through their agreement with measurements. At several plants this calculation of the DR is performed routinely before the start-up of the reactor after refueling, to avoid the possibility of instability during start-up (Sunde, 2007). The third form, and the most common, refers to determine the DR quantitatively with the relationship between two consecutive maxima of the autocorrelation function of the neutron noise. Despite being the form most used in the detection of instability in BWR, this has been questioned (van der Hagen, 2000), but still is the most used in practice.

A brief review of the second-order systems will give us the concepts needed to assess the stability of a system in general. A second order system can usually be expressed as follows:

$$\ddot{y}_{(t)} + 2\alpha\dot{y}_{(t)} + (\alpha^2 + \omega^2)y_{(t)} = 0 \quad (1)$$

The general solution for the system is:

$$y_{(t)} = Ae^{-\alpha t}\cos(\omega t + \phi) \quad (2)$$

The DR parameter gives a measurement of the dumping of the system and it is defined as the ratio between two consecutive maximums of the signal. For the second order system this parameter is a constant, and is given by:

$$DR = e^{-\frac{2\alpha\pi}{\omega}} \quad (3)$$

Additionally (see Fig. 2), a simpler definition is the average between two successive decay ratios (Kishida, 1990) measured between individual crests:

$$DR = \frac{A_{j+1}}{A_j} \quad (4)$$

Where A_j is the amplitude of the ith peak

Obtaining this parameter helps us study the stability of the system due to the following characteristics:

1) If the DR <1, the system is asymptotically stable (case of the Fig. 2).
2) If DR = 1, the system is critically stable.
3) If the DR> 1, the system is unstable

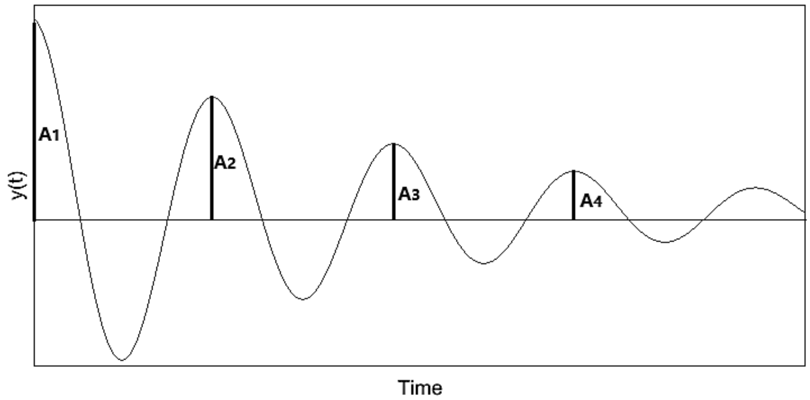

Figure 2. Decay Ratio definition.

This concept is not exclusive to second order systems, but can be extended to higher order systems whose response is oscillatory type, whether this is stable or unstable. For higher order systems overshoot can be obtained by analyzing the dominant poles of the system (Erbe, L., 1994).

4. METHODOLOGY OF ANALYSIS

In this work, the thermal-hydraulic system code RELAP5 and the 3D neutron kinetic code 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 nonlinear 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. Peach Bottom nodalization for RELAP5 and PARCS was based on the benchmark specification document for the turbine trip (TT) test (Solis et al., 2001) and on the data in the related tests report (Carmichael and Niemi, 1978). The Peach Bottom NPP core was divided into 33 heated regions representing the 764 real core fuel assemblies, modeled 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. Fig. 3 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. A recent study (Ambrosini and Ferreri, 2006) investigated stability boundaries obtained from a RELAP5 model for a boiling channel of 3.6 m with 48 and 24 meshes. The results showed that the stability boundaries predicted with 48 and 24 nodes are very similar. Therefore, the use of 24 meshes limits the complexity of the model reducing the calculation time and conserving the accuracy of the results.

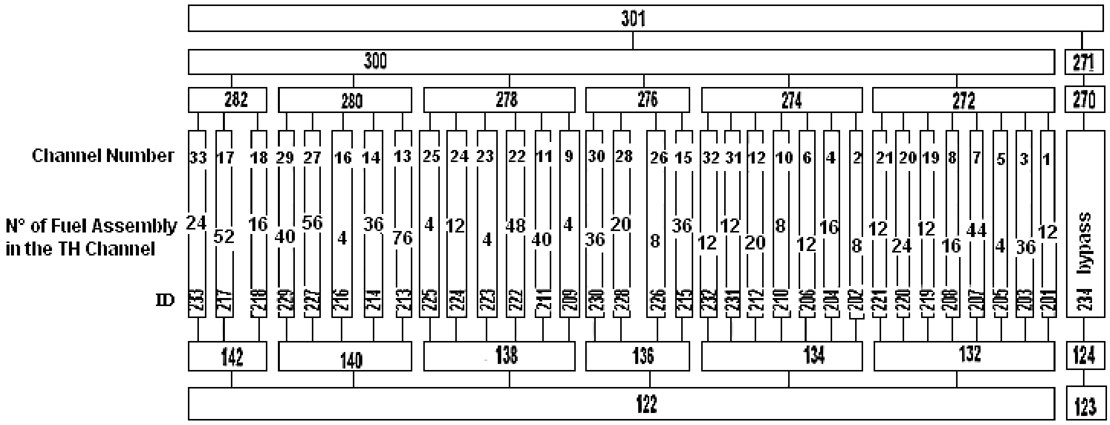

Figure 3. Part of the plant nodalization, with the 33 TH channels in the Peach Bottom reactor core.

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 1 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.

4. PRESSURE PERTURBATION EVENT

To analyze an event of instability, in this work a pressure pulse was generated in the turbine of the reactor; the amplitude of this pressure pulse is 0.22 MPa occurring in a time interval very short. Fig. 4 shows the peak pressure applied to the turbine in the PT3 operation point.

During the pressure perturbation (PP), a small pressure wave is created and it propagates along the steam line reaching the core zone through two main different paths: the steam separator filled with water and steam, and the lower plenum filled with water. When the wave reaches the vessel, it experiences several reflections with solid as well as fluid boundaries (Costa et al., 2007).

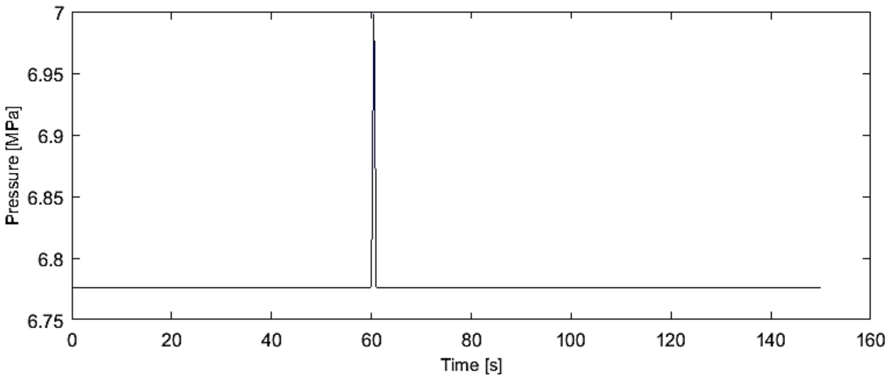

Figure 4. Pressure spike of 0.22 MPa applied in the turbine to simulate the pressure perturbation event.

5. RESULTS

5.1. Steady-state results

Before the analysis of the transient state, firstly, a steady-state analysis is performed in order to estimate the initial parameters of the thermal-hydraulic when the reactor is working before adding the pressure pulse; for this purpose, RELAP5 code was used independently, in this first estimate, it is considered that the axial power distribution is uniform and fixed. These initial conditions are then used to perform the calculations coupled.

In many cases, the parameter of most interest to be observed, in a steady-state calculation coupled, is the 3D spatial distribution of core power. However, the data measured in the reference document (Carmichael and Niemi, 1978) present only the axial mean power profiles. Therefore, the experimental axial mean power profiles are herein compared with the coupled calculation.

The average axial power distribution at steady state obtained is shown in Fig. 5, to the point PT3 stability test. The results of the calculations are outlined coupled compared with measured. As can be seen, the central axial middle powers and calculated measures are very close to each other.

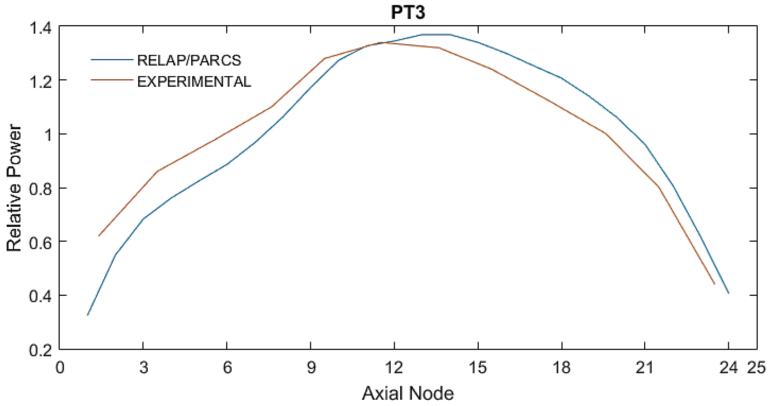

Fig. 5. PT3 experimental and calculated mean axial power profile.

Figure 6 shows the distribution of the radial average power to axial level 12. This distribution is considered fixed for this first analysis of the steady state. The PARCS code estimated with good precision the distribution for different axial levels of the core. Since the reference document (Carmichael, 1978) does not show the experimental radial distribution of power, it was not done a comparison. Fig. 6 will allow observing how changed the power distribution, once issued the pressure pulse.

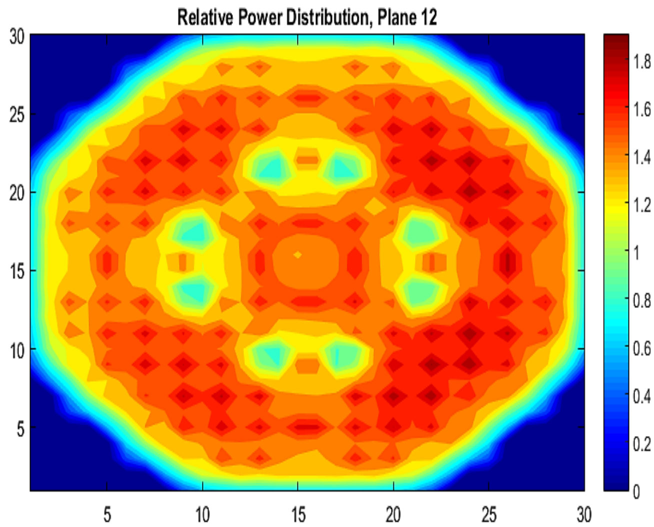

Figure 6. Calculated 2D steady state spatial core power distribution (axial level 12) PT3.

5.2. Transient results

The PP transient results for the point PT3 are presented in the Fig. 7 and 8. The figures show, respectively, the evolutions of power, void fraction (channel 1) and coolant temperature (channel 1). The perturbation begins at the time 60.5 s. The process presents a fast decrease in the power amplitude oscillation and, after about 43.0 seconds, oscillations are terminated. The system presents good stability to the PP transient. After the perturbation, the power, and the other parameters return to the steady state values.

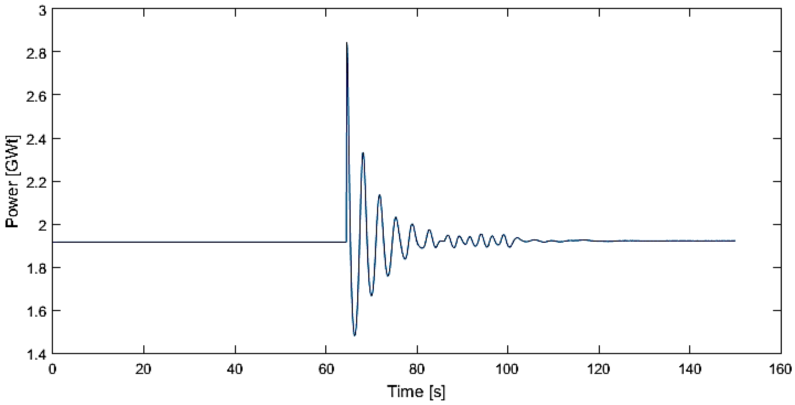

Figure 7. Power time evolution.

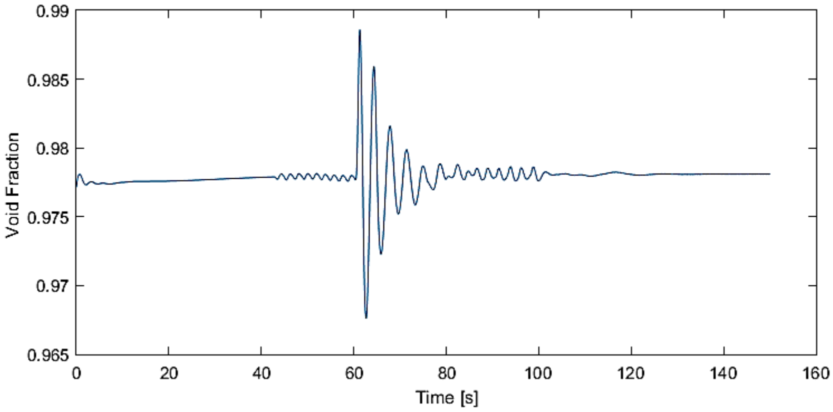

Figure 8. Void fraction time evolution at the axial level 12 in the channel 1.

Figures 7 and 8 show the behavior of the power and void fraction before and after the PP transient. In both figures , it how the system reacts almost immediately when the pressure pulse is emitted. As the immediate effect of the pressure perturbation in the core, the power increase abruptly and then it begins to decrease, in an oscillatory manner and finally converge to a constant value, the same as in the steady state operation. In the case of the void fraction, (Fig. 8) a similar behavior is observed. The oscillation of the values is maximized when the pressure pulse is emitted and then converges to a constant value. The fact that these variables converge to a constant value, returning to steady state conditions, shows that the system reaches stability after this specific transient event.

The variable of interest in the analysis of reactor stability is the change of power. For this reason, an analysis of power oscillation was done. In order to quantify the stability parameter, DR and NF were calculated. In this work, DR and NF were calculated using the MATLAB code and its tools; for this analysis it is considered that the power oscillation behaves as a second-order system and also the white noise is considered approximately null; under this condition, the DR and NF restrictions were obtained using Eq. (4) and (5).

The methodology followed was normalizing the behavior of the power (Fig. 9). After obtaining the normalized curve, with the tools of MATLAB, the peaks of the curve were located. Then, with these values, is made an adjustment in the behavior of the curve and finally DR and NF are calculated with the fitted curve. The values obtained are shown in Tab. 2 and they were compared with the results obtained using other calculations methodology, ADRI (Analysis of Decay Ratio Instability) and DRAT (Decay Ratio), described in (Costa et al., 2007).

The values obtained from DR and NF compared to those obtained by ADRI and DRAT are very close each other with a deviation of 6% for DR and 4.8% for NF using ADRI. ADRI model uses methods auto-regression (AR) to predict the value of DR and NF considering the noise of the signal. Furthermore, with DRAT model the deviations are very small, 3.8% in the case of DR and 3.1%. Now, comparing the results obtained with the DRAT model, the deviations are very small, 3.8% in the case of DR and 3.1% for NF.

Fig. 10 shows how the model obtained is very close to the original curve by which it can be assumed that the used model is a good approximation in relation to auto regression models. For obtaining the fitted curve, as with DR and NF, the Matlab code was used, this code by having a variety of tools facilitates obtaining the necessary parameters for adjustment; the "findpeaks" tool was used to locate the peaks of the signal, the values obtained are stored in the form of a matrix and these are considered as the maximum of the (Equation 2) function, once filtered the maximums, i.e. A_j, an adjustment is made on the solution of Eq. (1), always considered that is a second order differential equation.

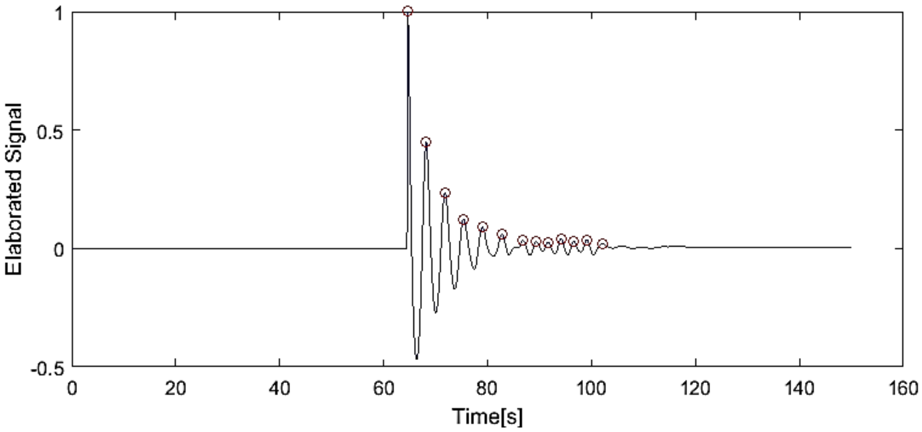

Figure 9. Normalized signal and the location of the peaks.

Table 2. DR and NF results in comparison with the experimental data for PT3 and ADRI and DRAT calculations.

Data from experimental results			**ADRI**		**DRAT**		**Model of This Work**	
LFST	DR	NF [Hz]	DR	NF [Hz]	DR	NF [Hz]	DR	NF [Hz]
PT3	0.344	0.437	0.454	0.289	0.467	0.275	0.485	0.284

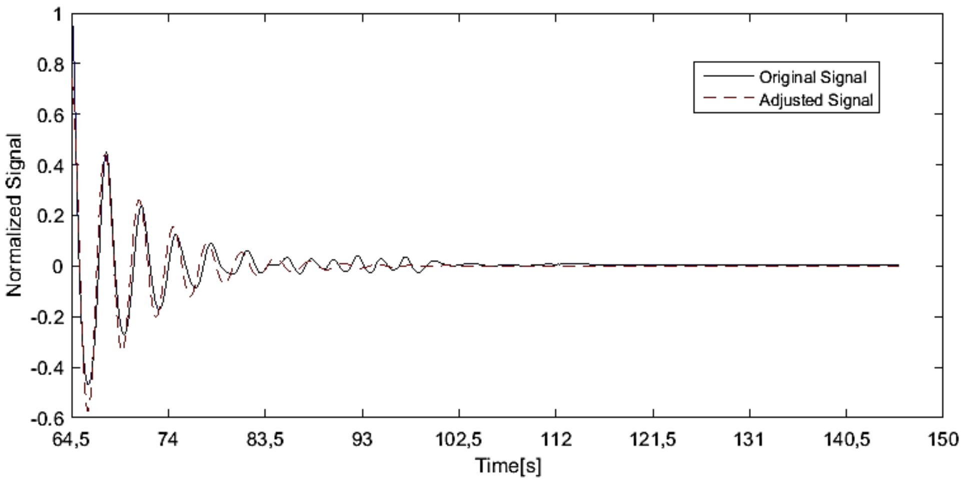

Figure 10. Original normalized signal and the fitted curve.

6. CONCLUSIONS

The transient event allowed analyzing the parameters of the stability DR and NF. The transient event consisted of an abrupt change of pressure in a turbine of a BWR reactor using a thermal hydraulic and neutronic coupled model. It is evident that this perturbation generates a variation in the behavior of the variables of the reactor, such as temperature, void fraction, mass flow rate, power, etc.

Fig. 8 showed how the void fraction oscillates once issued the pressure pulse; this variable may be one of the main reasons for which a BWR reactor falls to a region of instability. In this paper main attention was given to the analysis of power, and evolution after the disturbance. As it was seen in section 2, DR is a parameter to assess whether the reactor is in a stable region and there are many methods and models that calculate this parameter. The most widely used method is the model of auto regression AR and its variants such as ARMA, ARMAX and ARIMA that consider that the signals are affected by noises and these models are able to isolate the signal of interest, also predicting the behavior of the signal for future times.

The method used in this work came fairly accurately the methods of auto regression, but always bearing in mind that input signal behaves much like a system of second order and also noise should not be very significant; with these restrictions, it was possible to obtain the value of DR and NF very close to those obtained with ADRI and DRAT models. Figure 10 showed how the model obtained is very close to the original curve by which it can be assumed that the used model is a good approximation in relation to auto regression models.

7. ACKNOWLEDGEMENTS

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