Peach Bottom Cycle 2 Stability Analysis Using RELAP5 /PARCS

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ABSTRACT

Boiling channels and systems may oscillate owing to the behaviour of the liquid-steam mixture used for removing the thermal power. A thermal-hydraulic system may be unstable under particular operating conditions. Two kinds of power oscillation have been observed in BWR cores. One is an in-phase (core-wide) and the other is an out-of-phase (regional) oscillation. Since the above feature can make detection more difficult, the latter oscillation is potentially more severe. The problem is well known since the design of the first BWR system. However, to improve the safety systems of these reactors, it is necessary to be able to detect in a reliable way these oscillations from the neutronic signals.

The purpose of this work is to characterize the unstable behaviour of a BWR. Within this study, it has been performed a number of perturbation analysis. The coupled codes RELAP5-Mod3.3/PARCS have used for the simulation of the transients.

Validation has been performed against Peach Bottom-2 Low-Flow Stability Test PT3. Three dimensional time domain BWR stability analysis were performed on test point 3 for the core wide oscillation mode. In this transient dynamically complex events take place, i.e., neutron kinetics is coupled with thermal-hydraulics and an in-phase oscillation has been developed.

The calculated results are compared against the available experimental data.

1 INTRODUCTION

Experience has shown that, while single phase flows are normally stable, two phase flows (involving liquid and gas) may under certain conditions be prone to oscillatory behavior. Coupled neutronic-thermalhydraulic systems may show stable or unstable behavior: in the former case the effect of any disturbance occurring during a steady state condition is damped in time, in the latter case the disturbance is amplified and there is the possibility to reach self-sustained oscillating conditions, called “stable-limit-cycle”.

204.1
There are several types of thermalhydraulic instabilities which may occur also simultaneously in a boiling water reactor; each of these types can be classified by the appropriate physical mechanism or mode of oscillations.

In actual BWR operation, thermalhydraulic instability may be coupled with neutronic feedback.

Although no mechanism exists preventing the combination of the various identifiable instability modes, the thermalhydraulic density wave instability coupled with the neutronics feedback is commonly referred as the dominant mechanism triggering and sustaining instability in commercial BWRs.

The two modes of oscillation that are commonly recognized for density wave instabilities in a BWR plant are core wide oscillation and regional mode; these also referred as in-phase or out-of-phase mode respectively. In the core wide oscillation the power and inlet flow of the largest majority of core channels oscillate in-phase since they behave as a single channel. In the regional oscillation, the power of a region of the core oscillates out-of-phase with respect to the power of other regions. The inlet flows to the different regions are also out-of-phase with respect to each other. If only two halves of the core are involved, these behave as two parallel channels.

Oscillation in two phase systems may be connected with different mechanism related to pressure and density wave propagation, change in flow regime, interaction between conduction and convection heat transfer, coupling between thermal-hydraulic and neutronic parameters, presence of different parallel channels and of loops in parallel or in series with boiling channel.

Design parameters, like nominal pressure and pressure losses in single and two phase regions, can be properly selected to reduce the impact for the problem on reactor operation. However, the large variety of situations expected during the life of the core, also depending on the range of fuel burnup, requires a prudent analysis and the identification of a set of design parameters preventing the instability occurrence in most of possible BWR power plant operating conditions.

Both of above considerations testify of the complexity of the subject and give a reason why activities are still in progress.

So, there is the need to understand the effect of relevant parameters upon the involved physical phenomena, to detect these phenomena and to mitigate or suppress the eventual instability occurrences, using the safety margins adopted in the design.

In the recent years the need was felt to address the study of transient strongly characterized by neutronics-to-thermal-hydraulics feedbacks. The incorporation of full three-dimensional (3D) modelling of the reactor core into system transient codes allows “best estimate” simulations of interactions between reactor core behaviour and plant dynamics.

This paper deals with perturbation analyses in the Peach Bottom 2 Low Flow Stability Test point 3 conditions. This test was performed in the right boundary of the instability region of the Power/Flow Map, i.e. in the area of low flow (around 38% core flow rate) and high power (59.2%). The analysis has been carried out with the coupling code RELAP5mod3.3/PARCS.

2 ADOPTED CODES AND NODALISATIONS.

The reference Nuclear Power Plant considered in the analyses is the Unit-2 of Peach Bottom (General Electric designed Boiling Water Reactor), 3293 MWth power.

The core and neutronic data used in all the calculations are specified in [1].

2.1 Used Codes.

A thermal-hydraulic code (RELAP5-Mod3.3) and a 3D neutronic code (PARCS) have been adopted in this work.
In the calculation, the PARCS code uses the moderator temperature and density and the fuel temperature calculated by RELAP5 to incorporate appropriate feedback effects into the cross sections. Likewise, RELAP5 takes the space-dependent power calculated in PARCS and solves the heat conduction in the core heat structures. The time coupling between both codes is managed by the PVM library.

2.2 Nodalisations.

For all the calculations, it has been developed the same detailed thermal-hydraulic nodalisation reproducing each geometrical zone of the plant (Fig. 1, 2). The general methodology followed in setting up the nodalisation is described in [2].

For the core nodalisation one heated channel has been modelled to represent the active part of the core and one channel for all by-passes. For the rest of the plant a coarse nodalisation has been adopted for limiting the needed computer resources. A core nodalisation composed with 33 thermalhydraulic channels has also been developed in order to investigate the effect of the different number of T/H channels on the results (Figure 2 refers to this nodalisation).

For the neutronic code a nodalisation with a 3D core mesh composed with 764 nodes has been modeled [3]. A large set of cross section data including 435 compositions has been adopted in neutronic input deck [1].

![Fig. 1 - Core nodalisation](image1)

![Fig. 2 - Whole loop nodalisation](image2)
3 CALCULATION MODELS AND RESULTS.

The reactor operating conditions at which the test has been conducted are listed in Table 1[4].

<table>
<thead>
<tr>
<th>Reactor Power (Mwt)</th>
<th>Core Flow Rate (lb/h; kg/s)</th>
<th>Core Pressure (psia; Pa)</th>
<th>Core Inlet Entalphy (Btu/lb ; kj/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Value</td>
<td>1948</td>
<td>38.9 e+6</td>
<td>4907.7</td>
</tr>
<tr>
<td>% Rated</td>
<td>59.2</td>
<td>38.0</td>
<td>1005; 6.929e+6</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>528.4; 1229.058</td>
</tr>
</tbody>
</table>

3.1 Calculation procedures

Have been performed several perturbation analyses. The analyses begin always with a steady state calculation followed by a steady state coupled calculation and by a transient null calculation.

3.2 Steady state and null transient.

The steady state has been run for about 100 seconds, till thermal-hydraulic parameters of the whole loop reach stable temporal trends.

In the steady state coupled calculation with RELAP5/PARCS (second step) are sufficient around 9 seconds during which kinetics and thermal-hydraulic parameters converge toward stable values. After that, a 100 seconds null transient run have been set, to make it sure that the stable condition exists. Good agreement between calculated and experimental data is obtained. The results achieved are showed in the following Table and in Fig. 3:

Table 2 - Reactor main parameters prior to the disturbances

<table>
<thead>
<tr>
<th>Parameters, Units</th>
<th>Measured</th>
<th>Calculated (1 channel)</th>
<th>Calculated (33 channels)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Thermal Power, MWt</td>
<td>1948.0</td>
<td>1949.0</td>
<td>1949.0</td>
</tr>
<tr>
<td>Reactor Flow, kg/s</td>
<td>4907.7</td>
<td>5032.0</td>
<td>4525.077</td>
</tr>
<tr>
<td>Core Inlet Temperature, K</td>
<td>544.1</td>
<td>556.0</td>
<td>557.22</td>
</tr>
<tr>
<td>Core Inlet Enthalpy, kJ/kg</td>
<td>1229.058</td>
<td>1251.6</td>
<td>1258.1</td>
</tr>
<tr>
<td>Pressure at Core Outlet, Pa</td>
<td>6.929234e6</td>
<td>6.84919e6</td>
<td>6.937826e6</td>
</tr>
</tbody>
</table>

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3.3 Transients Calculations and Results.

Seven perturbation cases have been analyzed. The analyses carried out and the corresponding results are presented below:

Case a: Two Peaks Pressure Perturbation.

The transient calculation starts at 124 seconds; the reactor are disturbed with two peak pressure perturbations of around 0.06 MPa in the steam line (see Fig. 4).

The figure 5 shows the total reactor power evolution during this transient.
The disturbance in the steam line ends at 129 seconds and how it is clear from the Figure the perturbation doesn’t produce any unstable behavior in the reactor: after the conclusion of the disturbance, the power oscillation decreases very rapidly, becoming negligible in few seconds.

**Case b: Two Peaks Pressure Perturbation with Feedwater Flow Reduction.**

The pressure perturbations are the same of the Case a; in this analysis has also been adjoined a perturbation in the feedwater: at 100 seconds the feedwater flow has been reduced to a value around 280 kg/sec smaller.

The figure 6 represents the power trend for this case. The feedwater flow reduction generates a longer power oscillation compared to the Case a: as we can see in the figure, the oscillation dies out in approximately 10 seconds with a frequency of around 0.31 Hz and with an amplitude of 30 MW.

**Case c: Pseudo Random Binary Sequence Pressure Perturbation.**

The transient calculation begins at 120 seconds with a perturbation in the steam line constituted of fifty pressure peak obtained with a random function (see Fig.7).
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In figure 8 the total reactor power behavior during the transient has been plotted.

As it’s clear considering the figure, the system doesn’t develop an unstable behavior and, after the end of the perturbation, the power goes back very quickly to the initial condition.

**Case d:** Feedwater Temperature Reduction.

In this analysis the reactor is disturbed with a sudden variation of the feedwater temperature: at 129 seconds the temperature has been dropped 100 K.
The figure 9 presents the total reactor power trends:

![Graph showing total reactor core power in Case d.]

From the figure can be determined that the reactor shows an unstable behavior: 60 seconds after the end of the perturbation, the power grows up to 2590 MW and is developed an oscillation with a frequency of 0.35 Hz and with an amplitude that, at the beginning is of 230 MW, after reaches 560 Mw for stabilizing at a value of about 200 MW.

**Case e: Two Peak Pressure Perturbation with a Modified Axial Power Distribution.**

At the beginning of the transient the axial power distribution has been modified with a control rod movement: in 5 seconds some control rods groups have been withdrawn (the final situation is illustrated in Fig 10); then, the same pressure perturbation used in the Case a has been applied.

![Control rod movement diagram.]

In figure 11 is shown the change of the reactor axial power shape after the control rod movement.
In consequence of the new axial profile, the reactor parameters reach a new stable condition: the core inlet flow rate drops 280 Kg and the reactor power goes up to about 2010 MW.

In Fig. 12 and 13 are represented, respectively, the power and the core inlet flow rate evolution during the entire transient.

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Fig. 11 - Axial Power Distribution

Fig. 12 - Total Reactor Core Power in the Case e

Fig. 13 - Core Inlet flow rate in the Case e
With the new axial power distribution the perturbation, that in the former analyses doesn’t give satisfactory results, produces a regular oscillation behavior that maintains in the time with a constant frequency of 0.34 Hz and an amplitude of 100 MW for the power and of 80 kg/s for the Core Inlet flow rate.

Figure 14 emphasizes the regular oscillation power trend. We can observe that the oscillation reaches the limit cycle conditions.

![Figure 14 - Particular of the Power Oscillation in the Case e](image)

**Fig. 14 - Particular of the Power Oscillation in the Case e**

**Case f: Comparison Between Two Different Core Thermal-hydraulic Nodalisation.**

The same analysis described in the Case a has been made but adopting a different thermal-hydraulic nodalisation: to model the core, instead of using 1 channel, this analysis incorporate 33 thermalhydraulic channels.

Figure 15 shows a comparison between the trends calculated with the two different core thermalhydraulic models.

Observing the figure, we can note that the differences between the obtained results using the two distinct core nodalisations, are not very significant. Then, in order to reduce the CPU time, it is convenient and faster to use one channel model core.

![Figure 15 - Comparison between the results obtained with the two different models](image)

**Fig. 15 - Comparison between the results obtained with the two different models**
Case g: Comparison between Two Different Integration Method.

This case has been performed with the same perturbation of the Case a, however, the analysis has been computed with a different integration method. In the former transients semi-implicit integration method was used. This case has been analyzed with the nearly-implicit method.

Figure 16 represents the integration method effects on the total reactor power evolution.

Fig. 16 - Comparison between the results obtained with the two different computational methods

The perturbation produces similar results with the two different computational methods; however, it is to observe that, with the nearly implicit integration method, the reactor behavior appears slightly more unstable.

In the following table, frequencies and DR of the analyzed cases have been resumed and compared with the experimental data [4]:

Table 3 - Results of analyzed cases and comparison with measured data

<table>
<thead>
<tr>
<th></th>
<th>Frequency (Hz)</th>
<th>DR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case a</td>
<td>0.308</td>
<td>0.358</td>
</tr>
<tr>
<td>Case b</td>
<td>0.311</td>
<td>0.396</td>
</tr>
<tr>
<td>Case c</td>
<td>0.322</td>
<td>0.382</td>
</tr>
<tr>
<td>Case d</td>
<td>0.353</td>
<td>0.882</td>
</tr>
<tr>
<td>Case e</td>
<td>0.341</td>
<td>0.893</td>
</tr>
<tr>
<td>Case f</td>
<td>0.302</td>
<td>0.333</td>
</tr>
<tr>
<td>Case g</td>
<td>0.300</td>
<td>0.437</td>
</tr>
<tr>
<td>Measured</td>
<td>0.430</td>
<td>0.331</td>
</tr>
</tbody>
</table>
4 CONCLUSIONS.

Point 3 of the Low Flow Stability Tests performed at Peach Bottom NPP is a nearly stable point at the end of the cycle 2. This point is close to the stability boundary in the Power/ Mass Flow map, and besides, its axial power profile is not bottom peaked.

Nevertheless, with the analyzed cases, the characteristics of the in-phase instability can be recognized; for example, frequencies in all the oscillations produced in the analyses were from 0.3 to 0.4 Hz, i.e., in the typical frequency range of this kind of instability events.

In Case e (two peak pressure perturbation with modified axial power distribution) the reactor reaches the limit cycle condition. This analysis shows that the axial power shape affects the instability: the reactor developed an unstable behavior with regular oscillating trend only after the control rod movement, i.e., after the axial power distribution assumes a bottom peaked shape.

The situation simulated in the Case d is an extreme situation that is unlikely to occur in a real plant, so analyses with lower reduction of the feedwater temperature are programmed.

The comparison made in the Case g demonstrates that using 1 or 33 channels yields similar results so, it’s allowed to adopt a model with only one channel in order to reduce the CPU time, conserving the same accuracy.

REFERENCES.