ABSTRACT

The purpose of this thesis is the study of the thermalhydraulic oscillations in BWRs, where a strong non-linear coupling exists between the neutronic and thermalhydraulic processes via the void feedback reactivity. The objective is to contribute to understand the power oscillating conditions and to improve the methods capable to detect and describe these phenomena.

The reference design for the BWR is derived from data related to the Peach Bottom-2 BWR/4 Nuclear Power Plant.

Three dimensional time-domain BWR stability analyses have been performed for test point 3 (PT3) of the Low Flow Stability Tests carried out at Peach Bottom-2 during the first quarter of 1977 (at the end of cycle 2).

In the aim to better understand the instability development process, the stability response of the system around this operational point to several types of disturbances has been studied with the coupled codes RELAP5 Mod3.3/ PARCS, obtaining realistic and meaningful information on the reactor behaviour at the stability boundary in the Power/Flow Map.

In order to compare the results achieved with different thermalhydraulic nodalizations, all the transient analyses have been performed with two different models.

However, the most important and innovative contribution of this study is certainly the use, for the first time, of the data provided by the RELAP5/PARCS transient calculations to perform modal analyses with the VALKIN code, with very satisfactory results: with the coupled RELAP5 Mod3.3 and PARCS codes, detailed information regarding the status of the reactor it has been obtained as a function of time: mainly, the power distribution and the nuclear cross-sections for each core nodes; using these values, for all the disturbance tests, a power modal analysis was performed by the VALKIN code, with the aim to compare the power evolution obtained using a classical neutronic-thermalhydraulic coupled code and a modal code.

Moreover, in order to characterize the considered transient instabilities as "in-phase" or "out-of-phase" and also to study the relative importance of different modes during the transients, the oscillations of the power signals have been decomposed into its component modes.

For two perturbation tests, the results of the power modal decomposition have been also complemented with the information provided by the simulation of the LPRM signals by RELAP5/PARCS coupled codes: using data of the stable conditions of the system achieved from a steady state VALKIN calculations, a modal decomposition was performed of the neutronic power distribution obtained from the local power distribution in the reactor core (LPRM's signals from one of the axial level simulated in the RELAP5/PARCS transient calculation) and the information obtained from this decomposition was compared with the one available from the LPRM also simulated by these same coupled codes a very good agreement with the results of the modal decomposition performed using the nuclear cross-sections provided by the RELAP5/PARCS transient calculation has been demonstrated.

Then, for each perturbation test, the Decay Ratio and the Natural Frequency of the reactor have been calculated and the phase shift of LPRM signals located in opposite reactor zones (given by the RELAP5 Mod3.3/PARCS calculations) was analysed in order to examine the characteristics of the oscillations developed.

Finally, to investigate the effect of the use of distinct thermalhydraulic–neutronic coupled codes, it has been performed a perturbation analysis also with the coupled codes TRAC-BF1/VALKIN.

CONTENTS

ABSTRACT 1

CONTENTS 3

LIST OF FIGURES 7

LIST OF TABLES 11

1 INTRODUCTION 12

2 INSTABILITY IN BWR NPPS 21

2.1 GENERALITIES 22
2.1.1 Basic Phenomenology and historical perspective 22
2.1.2 Technical considerations 23
2.1.3 Safety relevance 24
2.2 PHENOMENOLOGY OF BWR PLANT INSTABILITY 26
2.2.1 Qualitative evaluation 26
2.2.2 Classification of instabilities 29
2.2.2.1 Classification of density wave instability 31
2.2.2.2 Physical mechanism of density wave oscillation with neutronic feedback 32
2.2.2.3 Relevance of basic phenomena to BWR technology 34
2.2.2.4 Parameters affecting density wave oscillations with neutronic feedback 35
2.3 MODELING FEATURES AND ASSESSMENT 36
2.3.1 Basic models 37
2.3.1.1 General description of available modeling techniques 37
2.3.1.2 Models for linear stability analysis 38
2.3.1.3 Models for describing non-linear behaviour 39
2.4 PLANT MONITORING, PREVENTION AND MITIGATION OF INSTABILITIES 40
2.4.1 Instrumentation capabilities 41
2.4.2 Neutron flux measurements 41
2.4.2.1 Measurement of other physical parameters 42
2.4.3 Use of the capability of the monitoring system 43
2.4.4 Control systems 46
2.4.4.1 Power/Flow map 47
2.4.4.2 Power limitation and protection system 47
2.4.5 Means to measure and quantify the stability of reactors 48
2.4.5.1 Effect of instability on process parameters 48
2.4.5.2 Stability monitors 49
2.4.6 Current strategies for prevention and mitigation of instability 52
2.4.6.1 Design aspects 52
2.4.6.2 Implemented strategies 53
2.4.6.2.1 Regional exclusion *53*
2.4.6.2.2 Quadrant APRM *53*
2.4.6.2.3 Power reduction derived from recirculation pump trip or loss- of feedwater preheater *54*

3 PLANT DESCRIPTION 55

3.1 GENERAL PLANT DESCRIPTION 56
3.2 CORE GEOMETRY AND FUEL ASSEMBLY 62
3.3 NUCLEAR INSTRUMENTATION DATA 68

4 DESCRIPTION OF ADOPTED CODES 70

4.1 RELAP5/ MOD 3.3 CODE 71
4.1.1 Development of RELAP5/MOD 3.3 72
4.1.2 Code architecture 73
4.1.3 Hydrodynamic model....76
4.1.4 Heat structure models 79
4.1.5 Trip system....81
4.1.6 Control system....82
4.1.7 Special techniques....82
4.2 PARCS 2.4 CODE....83
4.2.1 Introduction 83
4.2.2 Calculation features 85
4.2.2.1 Eigenvalue calculation 85
4.2.2.2 Transient (kinetics) calculation....85
4.2.2.3 Xenon transient calculation....86
4.2.2.4 Decay heat calculation 86
4.2.2.5 Pin power calculation....86
4.2.3 Modeling features 87
4.2.3.1 Geometric representation 88
4.2.3.2 Cross-section functionalization....88
4.2.3.3 Thermalhydraulics feedback 89
4.3 RELAP5/PARCS COUPLING....90
4.3.1 Features of the RELAP5/PARCS codes 90
4.3.2 The MAPTAB file 91
4.3.3 Calculation methodology 93
4.4 VALKIN CODE....95
4.4.1 Introduction 95
4.4.2 Nodal modal method....95
4.4.2.1 Modes updating....98
4.5 SIGNAL MODAL DECOMPOSITION 99
4.6 TIME SERIES ANALYSIS....101
4.6.1 Basic definitions 101
4.6.1.1 Analytical definition of DR 102
4.6.1.2 Autocorrelation function....103
4.6.1.3 Parametric model 103
4.6.2 Definitions of the Decay Ratio 106
4.6.2.1 Standard method (DR1 and DR2)....106
4.6.2.2 AR-Lyapunov approximation (DR3) 108
4.6.2.3 ARMA-AR model (DR4) 109
4.6.3 Decay Ratio calculations....110

5 TRANSIENTS DESCRIPTION:THERMALHYDRAULIC AND NEUTRONIC MODELING 111

5.1 TRANSIENTS DESCRIPTION....112
5.1.1 Peach Bottom-2 Low Flow Stability Tests 112
5.1.1.1 Plan of the experiment 113
5.1.1.2 Actual Test conditions 114
5.1.1.3 Test procedures....118
5.2 TRANSIENTS CALCULATIONS 119
5.2.1 Case A: Two Peak Pressure Perturbation....120
5.2.2 Case B: Pseudo Random Sequence Pressure Perturbation 120

5.2.3 Case C: Two Peak Pressure Perturbation with a Modified Axial Power Distribution 121
5.3 DATA AVAILABILITY 123
5.4 THERMALHYDRAULIC MODELLING 123
5.4.1 Recirculation loop 128
5.4.2 Steam line 130
5.4.3 Feedwater line 131
5.4.4 Reactor vessel 131
5.4.5 Adopted core nodalizations 132
5.5 NEUTRONIC MODELLING 136
5.5.1 BWR LPRM response model 140
5.6 3-D NEUTRONIC/THERMALHYDRAULIC COUPLING 142

6 RESULTS 143

6.1 CALCULATION STEPS 145
6.1.1 RELAP5/PARCS Coupled Calculation 145
6.1.2 Decay Ratio and Natural reactor Frequency Calculation 145
6.1.3 Power modal analysis 146
6.1.4 Decomposition of the LPRM signals 147
6.2 STEADY STATE RESULTS 148
6.3 TRANSIENTS RESULTS 150
6.3.1 Case A: Two Peaks Pressure Perturbation 150
6.3.1.1 Case A1: Two Peaks Pressure Perturbation (model with 33 core channels) 150
6.3.1.2 Case A2: Two Peaks Pressure Perturbation (model with 1 core channel) 157
6.3.1.3 Case A: Two Peaks Pressure Perturbation (comparison between the results achieved with the two different nodalizations) 160
6.3.2 Case B: Pseudo Random Sequence Pressure Perturbation 161
6.3.2.1 Case B1: Pseudo Random Sequence Pressure Perturbation (model with 33 core channels) 161
6.3.2.2 Case B2: Pseudo Random Sequence Pressure Perturbation (model with 1 core channel) 164
6.3.2.3 Case B: Pseudo Random Sequence Pressure Perturbation (comparison between the results achieved with the two different nodalizations) 168
6.3.3 Case C: Two Peaks Pressure Perturbation with a modified axial power distribution 170
6.3.3.1 Case C1: Two Peaks Pressure Perturbations with a modified axial power distribution (model with 33 core channels) 170
6.3.3.2 Case C2: Two Peaks Pressure Perturbation with a modified axial power distribution (model with 1 core channel) 182
6.3.3.3 Case C: Two Peaks Pressure Perturbation with a modified axial power distribution (comparison between the results achieved with the two different nodalizations) 188
6.3.4 Oscillation analysis: Decay Ratio and Natural Reactor Frequency calculation and analysis of the LPRM signals 191
6.4 COMPARISON BETWEEN THE OBTAINED RESULTS WITH TWO DIFFERENT THERMALHYDRAULIC-NEUTRONIC COUPLED CODES 199
6.4.1 TRAC/BF1-VALKIN code 199
6.4.1.1 TRAC/BF1-VALKIN coupling 200
6.4.2 Transient Description and thermalhydraulic and neutronic modeling 201
6.4.3 Comparison of results 201
6.4.3.1 Steady state results 201
6.4.3.2 Transient results 202

7 CONCLUSIONS 204

7.1 OBSERVATIONS OF THE OBTAINED RESULTS 205
7.2 CONSIDERATIONS ON THE ADOPTED CODES 207
7.3 AREAS OF THE FUTURE INVESTIGATION 211
7.4 PUBLICATIONS 212

REFERENCES **213**

LIST OF FIGURES

Figure 2.1: Example of possible idealization of thermalhydraulics and neutronics feedback following density waves in BWR plants 34
Figure 2.2: Procedure followed in linear stability analysis of BWRs 39
Figure 2.3: Classification of non-linear models adopted in the stability analysis of BWRs 40
Figure 2.4: Typical BWR operating Power/Flow map (the thick lines identify the region not allowed for normal operation). 46
Figure 3.1: Peach Bottom-2 reactor vessel and internals 57
Figure 3.2: Peach Bottom-2 recirculation system 60
Figure 3.3: Peach Bottom-2 main steam lines system 62
Figure 3.4: Radial distribution of assembly types 65
Figure 3.5: Assembly type identification, TIP and Control rod location 66
Figure 3.6: Core orificing and TIP system arrangement 68
Figure 3.7: Elevation of core components 69
Figure 4.1: RELAP 5 top level structure 73
Figure 4.2: Modular structures of transient calculation in RELAP 5 75
Figure 4.3: Transient/Steady state block structure 75
Figure 4.4: Association between thermalhydraulic and neutronic nodalization ... 92
Figure 4.5: RELAP5/PARCS external coupling calculation methodology 94
Figure 4.6: Adjacent nodes to LPRM 101
Figure 4.7:Definition of the DR 107
Figure 5.1: Peach Bottom-2 Low Flow Stability Tests. planned test conditions 113
Figure 5.2: Peach Bottom-2 EOC 2 test planned test operational time line 114
Figure 5.3: Peach Bottom-2 Low Flow Stability Test PT3 control rod pattern .. 116
Figure 5.4: Peach Bottom-2 Low Flow Stability Test PT3 average axial power distribution 116
Figure 5.5: Peach Bottom-2 Low Flow Stability Tests. actual test conditions ... 117
Figure 5.6: Two peaks pressure perturbation 120
Figure 5.7: Pseudo Random Pressure Perturbation 121
Figure 0.8: Control rods movement 123
Figure 5.9: a) RELAP5 nodalization for the core modelled by a single T/H channel; b) correspondence between core thermal hydraulic model (1 channel) and 3-D neutron kinetics nodalization 124
Figure 5.10: RELAP5 nodalization for the core modeling by 33 T/H channels . 125
Figure 5.11: RELAP5 nodalization of Peach Bottom-2 126
Figure 5.12: Recirculation loop schematization 129
Figure 5.13: Jet pump schematization 130
Figure 5.14: Simplified sketch of the steam line 131
Figure 5.15: Identification of the 33 thermalhydraulic channels 133
Figure 5.16: Radial Distribution of Assembly Types 137
Figure 6-1: Comparison of RELAP5/PARCS calculated core average axial power distributions with experimental test (process computer corrected). 149
Figure 6-2: Relative core power variation during the transient for Case A1 151
Figure 6-3: Relative core inlet mass flow rate variation during the transient for Case A1 151
Figure 6-4: Relative steam line pressure variation during the transient for case A1 152

Figure 6-5: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with different number of eigenvalues. 153
Figure 6-6: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 154
Figure 6-7: Power evolution with RELAP/PARCS and VALKIN, and amplitudes of the first three modes.. 155
Figure 6-8: Amplitude evolutions obtained from the LPRM signals for case A1 156
Figure 6-9: Comparison between the amplitudes obtained in the two distinct decompositions and the power evolution achieved with the RELAP5/PARCS calculation 156
Figure 6-10: Relative core power variation during the transient for case A2..... 157
Figure 6-11: Relative core inlet mass flow rate variation during the transient for case A2 158
Figure 6-12: Relative steam line pressure variation during the transient for case A2 158
Figure 6-13: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 159
Figure 6-14: Particular of the power evolution comparison plotted in the figure 6.13 159
Figure 6-15: Comparison between the relative core power trends obtained with the two different models 160
Figure 6-16: Comparison between the relative core inlet mass flow rate transient evolutions obtained with the two different models 160
Figure 6-17 Comparison between the relative steam line pressure histories during the transient obtained with the two different models.. 161
Figure 6-18: Relative core power evolution during the transient for case B1 162
Figure 6-19: Relative core inlet mass flow rate evolution during the transient for case B1 162
Figure 6-20: Relative steam line pressure evolution during the transient for case B1.. 163
Figure 6-21: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 164
Figure 6-22: Particular of the comparison plotted in figure 6.21: transient evolution between 10 sec and 60 sec 164
Figure 6-23: Relative core power trend during the transient for case B2........... 165
Figure 6-24: Relative core inlet mass flow rate trend during the transient for case B2.. 165
Figure 6-25: Relative steam line pressure trend during the transient for case B2.. 166
Figure 6-26: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 166
Figure 6-27: Particular of the comparison plotted in figure 6.26: transient evolution from 10s to 60 s.. 167

Figure 6-28: Particular of the comparison plotted in figure 6.26: transient evolution from 30s to 53s 167
Figure 6-29: Comparison between the relative core power trends obtained with the two different models 168
Figure 6-30: Particular of the core power trends comparison plotted in the preceding figure: power histories from the end of the perturbations to the end of the transient 169
Figure 6-31: Comparison between the relative core inlet mass flow rate transient evolutions obtained with the two different models 169
Figure 6-32: Comparison between the relative steam line pressure histories during the transient obtained with the two different models........ 170
Figure 6-33: Comparison between the axial power distributions calculated before and after the control rod movement in case C1 and the reference one........ 171
Figure 6-34: Reactor core power in case C1 171
Figure 6-35: Particular of the power oscillation in case C1........ 172
Figure 6-36: Core inlet mass flow rate in case C1 172
Figure 6-37: Particular of the core inlet flow rate oscillation in case C1 173
Figure 6-38: Steam line pressure in case C1 173
Figure 6-39: Particular of the steam line pressure oscillation in case C1........ 174
Figure 6-40: Particular of the reactor core power history during the transient in case C1 175
Figure 6-41: Reactor core power trend during the perturbation in case C1........ 175
Figure 6-42: Particular of the reactor core inlet mass flow rate history during the transient in case C1 176
Figure 6-43: Reactor core inlet mass flow rate trend during the perturbation in case C1 176
Figure 6-44: Particular of the steam line pressure history during the transient in case C1 177
Figure 6-45: Steam line pressure during the perturbation in case C1........ 177
Figure 6-46: : Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 178
Figure 6-47: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 3 eigenvalues without updating the process. 179
Figure 6-48: Amplitude evolutions obtained from the LPRM signals for case C1 180
Figure 6-49: Comparison between the amplitude evolution associate with the fundamental mode and the power trend achieved with the RELAP5/PARCS calculation 181
Figure 6-50: 3-D representations of the average power evolution during the transient time in case C1: from 147.7s to 149.7s 182
Figure 6-51: Comparison between the axial power distributions calculated before and after the control rod movement in case C2 and the reference one........ 183
Figure 6-52: Reactor core power history during the transient in case C2 184
Figure 6-53: Core inlet mass flow rate during the transient in case C2........ 184
Figure 6-54: Steam line pressure history during the transient in case C2 185
Figure 6-55: Reactor core power evolution during the perturbation in case C2 . 185

Figure 6-56: Core inlet mass flow rate evolution during the perturbation in case C2 186
Figure 6-57: Steam line pressure evolution during the perturbation in case C2 . 186
Figure 6-58: Comparison between the power evolution obtained with RELAP5/PARCS and the power evolution obtained using VALKIN code with 1 eigenvalue and different updating times. 187
Figure 6-59: Particular of the comparison represented in the previous figure.... 187
Figure 6-60: Control rod movement comparison between the core power evolutions obtained with the two different thermalhydraulic models in case C . 188
Figure 6-61: Control rod movement comparison between the core inlet mass flow rate evolutions obtained with the two different thermalhydraulic models in case C 189
Figure 6-62: Control rod movement comparison between the steam line pressure evolutions obtained with the two different thermalhydraulic models in case C . 189
Figure 6-63: Comparison between the core power evolutions obtained with the two different thermalhydraulic models in case C during the disturbance 190
Figure 6-64: Comparison between the core inlet mass flow rate evolutions obtained with the two different thermalhydraulic models in case C during the disturbance 190
Figure 6-65: Comparison between the steam line pressure evolutions obtained with the two different thermalhydraulic models in the Case C during the disturbance 191
Figure 6-66 LPRM system arrangement 193
Figure 6-67: Comparison between opposite LPRM signals in case A1 194
Figure 6-68: Comparison between opposite LPRM signals in case A2 194
Figure 6-69: Comparison between opposite LPRM signals in case B1 195
Figure 6-70: Comparison between opposite LPRM signals in case B2 195
Figure 6-71: Comparison between opposite LPRM signals in case C1 196
Figure 6-72: Comparison between opposite LPRM signals in case C2 196
Figure 6-73: Comparison between opposite LPRM signals in case C1 197
Figure 6-74: Comparison between opposite LPRM signals (LPRM north and south) in case C1 198
Figure 6-75: Comparison between opposite LPRM signals (LPRM east and west) in case C1 198
Figure 6-76: Comparison between opposite LPRM signals (LPRM north and east) in the Case C1 199
Figure 6-77 Comparison of RELAP5/PARCS and TRAC-BF1/VALKIN calculated core average axial power distributions with experimental test (process computer corrected). 202
Figure 6-78: Comparison between RELAP5/PARCS and TRAC/VALKIN with no update and with updating time of 1 second. 203

LIST OF TABLES

Table 2-1: Summary classification of density wave instabilities 31
Table 2-2: Summary of main features of different approaches as a measure for prevention and suppression and mitigation of instabilities 44
Table 3-1: Peach Bottom-2 reference design information 58
Table 3-2: Reactor vessel design data... 59
Table 3-3: Reactor recirculation system design characteristics 61
Table 3-4: Peach Bottom-2 fuel assembly data.. 63
Table 3-5: Definition of assembly types 64
Table 3-6: Control rod data (movable control rod) 64
Table 3-7: Composition numbers in axial layer for each assembly type 67
Table 5-1: Interim technical specification Rod Block and APRM Scram Lines 114
Table 5-2: Peach Bottom-2 end of cycle 2 test actual Low Flow Stability Test conditions 115
Table 5-3: Main characteristics of the nodalization for the RELAP5 input deck with 33 channels 127
Table 5-4: Main characteristics of the nodalization for the RELAP5 input deck with 1 channel.. 128
Table 5-5: Recirculation system geometrical data 129
Table 5-6: Jet pump geometrical data... 129
Table 5-7: Steam line and bypass system geometrical data.............................. 131
Table 5-8: Reactor vessel geometrical data 132
Table 5-9: Data used for modelling 33 thermalhydraulic channels................... 135
Table 5-10: Range of variables... 138
Table 5-11: Key to macroscopic cross-section tables....................................... 139
Table 5-12: Macroscopic cross-section table structure..................................... 139
Table 6-1: Reactor main parameters prior to its disturbance 148
Table 6-2. Time series analyses results.. 192
Table 6-3- Reactor main parameters prior to its disturbance 202
Table 6-4 Time series analyses results... 202

1 INTRODUCTION

Experience has shown that, with respect to single-phase, two-phase flows (involving liquid and steam or gas) may more frequently be prone to oscillatory behaviour under particular conditions.

Coupled neutronic-thermalhydraulic systems may show stable or unstable behaviour: in the former case the effect of any disturbance occurring during a steady condition is damped in time, while in the latter case the disturbance is amplified and there is the possibility to reach self-sustained oscillating conditions, called “stable-limit-cycles”.

This is a well known drawback in boiling water technology that may complicate the very low pressure operation and is mitigated only at a conveniently high pressure. As such, the problem has been investigated since the start of the BWR technology: parameters affecting the stability were identified through the use of more or less sophisticate predictive models and computational tools. Proper countermeasures were taken at a design level, essentially keeping low the pressure drops in the two-phase region inside the core and downstream it, while increasing them in the single-phase region of the loop.

Over a period of several years there have been approximately thirty instability events in commercial BWRs. In-core reactor tests have been performed to study the stability behaviour and a few unplanned events occurred during normal operations, essentially start-up processes or recirculation pump trip transients. Then, the event in LaSalle-2 plant in March 1988 [1] that caused a high neutron flux scram attracted again the attention toward this topic. Since the US NRC issued notices and asked the BWR utilities to take a long term action to solve the stability problem, international interest on this topic has grown significantly. A wide review of reported instability events can be found in [2].

These instabilities were identified as periodic oscillations of the neutron flux detected via instrumentation readings. Essentially, neutronic power signals from local power range monitors (LPRM’s) and average Power Range Monitors (APRM’s) have been used to detect and study the power oscillations.

Oscillations 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 thermalhydraulic 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 of 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.

The 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 on the involved physical phenomena, to detect these phenomena and to mitigate or suppress the possible instability occurrences, using the safety margins adopted in the design.

There are several types of thermalhydraulic instabilities which may occur also simultaneously in a boiling water reactor; each of these types can be distinguished considering the particular physical mechanism or the mode of oscillation.

In this work the attention is concentrated on the instabilities that are known as neutronic-thermalhydraulic instabilities [3] and that are commonly referred to as the dominant mechanism triggering and sustaining instability in commercial BWRs.

In fact, in actual BWR operation, thermalhydraulic density wave instability may be coupled with neutronic feedback and there is no way of preventing the combination of the various identifiable instability modes.

The two modes of oscillation that are commonly recognized for density wave instabilities in a BWR plant are core wide and regional oscillations; 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 approximately 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.

Sophisticate models were set up and are still being developed by different organizations to respond to the needs of stability analyses. They are based on different approaches to the problem of stability and can be classically subdivided into the two classes of time-domain and frequency-domain codes. Codes in the former class are suitable for the non-linear analysis of the transient behaviour of BWRs during unstable conditions. On the other hand, frequency-domain codes have the capability to perform the linear stability analysis of such complex systems, supplying figures which quantify the margin to instability.

The possibility of instability in the core of a Boiling Water Reactor induced by thermalhydraulic and void reactivity feedback has been the subject of many analytical and experimental investigations. The result researches are often no directly applicable or extrapolated to BWR plants. The main reason is, generally, that the involved ranges of parameters, specifically geometry, pressure and type of fluid, are very different from those of concern to BWR cases.

However, these results may be used to qualify codes, to better understand basic phenomena, to stimulate research and to point out possible critical BWR plant situations. Parametric studies can also be carried out easily.

So, the main objectives of BWR stability analyses can be summarized as follows:

- to assess the stability margins in reactor plants, including normal and off-normal conditions;
- to predict the transient behaviour of the reactor, should unstable condition occur;
- to help in designing and to assess the effectiveness of the countermeasures adopted to prevent and mitigate the consequences of instability.

Such objectives can be obtained only through a realistic simulation of relevant physical phenomena and instability mechanism.

The purpose of this thesis is the study of the thermalhydraulic oscillations in BWRs, where a strong non-linear coupling exists between the neutronic and thermalhydraulic processes via the void feedback reactivity. The objective is to contribute to understand the power oscillating conditions and to improve the methods capable to detect and describe these phenomena.

In order to characterize the unstable behaviour of the BWR reactors, a number of perturbation analyses have been performed in relation to the Peach Bottom-2 Low Flow Stability Test point 3 (PT3) conditions.

Arrangements with Philadelphia Electric Company (PECo) were made by the Electric Power Research Institute (EPRI) in collaboration with General Electric company for conducting different series of Low Flow Stability Tests at Peach Bottom-2, during the first quarter of 1977.

The Low Flow Stability Tests were intended to measure the reactor core stability margins at the limiting conditions used in design and safety analysis, providing a one-to-one comparison to design calculations.

These tests were performed in the right boundary of the instability region in the Power/Flow Map, i.e. in the area of low flow (around 38% core flowrate) and high power (59.2%).

In the aim to better understand the process of instability development, the stability behaviour of this operational point (PT3) has been studied, carrying out a number of perturbation analyses with the coupled codes RELAP5 Mod3.3/ PARCS.

The coupling between these codes was not specifically designed to cope with BWR stability problem, but, since the resulting tool has general capabilities for a detailed thermalhydraulic and neutronic description of nuclear reactors, it can be used for analysing plant instability events with very satisfactory results, as shown in this thesis.

With the coupled RELAP5 Mod3.3 and PARCS codes, detailed information regarding the status of the reactor it has been obtained as a function of time: mainly, the power distribution and the nuclear cross-sections for each core nodes. Using these values, for all the disturbance tests, a signal modal decomposition was also performed by the VALKIN code (see Chapter 4), with the aim to compare the power evolution obtained using a classical neutronic-thermalhydraulic coupled code and a modal code.

Additionally, in order to characterize the addressed transient as "in-phase" or "out-of-phase" and also to study the importance of different modes during the transients, the oscillations of the power signals have been decomposed into its component modes.

In order to simulate realistic transients without calculate a large number of modes the nodal modal method implemented in the VALKIN code makes use of an updating process for the modes at certain time step. So, with the aim to observe the difference between the results obtained using different numbers of modes or different updating times, several transient calculations have been performed. The process of updating the modes increases considerably the accuracy of the obtained solution but is an expensive process from the computational point of view, thus it has been necessary to find a compromise between the number of modes and their updating frequency to optimise the performance of the method.

Moreover, for two perturbation tests, the results of the modal power decomposition have been complemented with the information provided by the simulation of the LPRM signals by RELAP5 Mod3.3/PARCS coupled codes; a modal decomposition was performed of the neutronic power distribution obtained from the local power distribution in the reactor core (made available by the coupled codes) and the information obtained from this decomposition was compared with the one available from the LPRM also simulated by these same coupled codes.

Finally, for each perturbation test, the Decay Ratio and the Natural Frequency of the reactor have been calculated and the phase shift of LPRM signals located in opposite reactor zones (given by the RELAP5 Mod3.3/PARCS calculations) was analysed in order to examine the characteristics of the oscillations developed.

The specific contributions of this thesis are shortly summarised hereafter.

- The most important and innovative contribution of this study is the use, for the first time, of the data provided by the coupled codes RELAP5 Mod3.3/PARCS to perform signal modal analyses with the VALKIN code with results very satisfactorily.

- Moreover, with this investigation, realistic and meaningful information was obtained about the reactor behaviour at the stability boundary of the Power/Flow Map, in addition to demonstrating that the small pressure perturbation tests offer an operationally simple and precise technique for determining BWR core stability margins.

- Other interesting results were obtained from the modal decomposition of the LPRM's signals simulated by the RELAP5 Mod3.3/PARCS transient calculations: using information of the stable conditions of the system achieved from the steady state VALKIN calculations it has been performed a modal decomposition of the neutronic power from the local power distribution in the reactor core (LPRM's signals from one of the axial level simulated in the RELAP5 Mod3.3/PARCS transient calculation) and it has been demonstrated a very good agreement with the results of the modal decomposition performed using the nuclear cross-section provided by the RELAP5 Mod3.3/PARCS transient calculation. This result is of great practical importance because demonstrates that, in theory, with this methodology it is possible in a nuclear plant to obtain on-line information concerning to the reactor stability.

- Finally, to investigate the effect of the use of distinct thermalhydraulic–neutronic coupled codes, it has been performed a perturbation analysis also with the coupled codes TRAC-BF1/VALKIN obtaining a very good agreement with the results achieved with RELAP5 Mod3.3/PARCS.

This document is organized as follows.

- The Chapter 2 describes the instability phenomena of interest for BWRs, addressing the relevant phenomenology, the physical mechanism, the codes available to study the occurrence of instabilities, the capabilities of instrumentation in monitoring the instability event and the current strategies for the prevention and the mitigation of instability:
  - parameters affecting the stability performance of BWR plants it has been identified and characterized;
  - a classification of the codes available for simulating, describing, and predicting instability phenomena is proposed;
  - methods of prevention and mitigation or suppression of instabilities in a BWR plant are described; so, instrumentation, plant control and protection systems, data interpretation and current strategies for prevention and mitigation are considered.

- Chapter 3 provides a description of the plant selected to perform the analyses: this chapter specifies also the core and neutronic data to be used in all the calculations.

- Chapter 4 describes the main characteristics of the RELAP5 and PARCS codes in order to show the way in which the plant modelling can be obtained, putting the bases to understand the nodalization described in the next chapter; the methods used for the signal modal decompositions and for the time series analyses are also described. The Chapter has been divided in five parts respectively concerning:
  - the thermalhydraulics (RELAP5);
  - the neutronics (PARCS);
  - the thermalhydraulics-neutronics coupling (RELAP5/PARCS coupled codes)

- the signal modal decomposition (VALKIN);
- the time series analysis.

- Chapter 5, in its first part, provides the specifications given and the options chosen to perform the analyses; in the second part dealt the description of the nodalizations developed for thermalhydraulic and neutronic modeling. This last part has been divided along the same lines of the first one of the chapter 4: a section for the thermalhydraulics, other one for the neutronics and one for the coupling between thermalhydraulic and neutronic.

- Chapter 6 presents the discussion of the results obtained from this work.

- Chapter 7 presents the conclusions and some recommendations for future work.

Some of the obtained results were processed to be presented by video clips of 3-D phenomena and are included into the attached CD-ROM.