7 CONCLUSIONS

The purpose of this thesis was to contribute to the study of the neutronic-thermalhydraulic instability in a BWR NPP. The point 3 test conditions of the Low Flow Stability Tests in the Peach Bottom reactor was chosen as real steady state conditions to start trying to approach the reactor stability boundary in the Power/Flow Map.

Several perturbation cases have been investigated with the objective to contribute to the understanding of the power oscillation conditions and to improve the methods to detect, describe and suppress these kind of phenomena.

Results of fundamental research might not be directly applicable or extrapolated to BWR plants; this is the case of the data presented in this thesis. The main reasons for this are, generally, that the involved ranges of parameters, specifically geometry, pressure and type of fluid, are very different from those of concern to BWR cases. Nevertheless, this type of analyses is very useful because it helps to better understand the studied phenomena, to qualify the adopted codes and to prevent and mitigate the consequences of instability events.

The most important conclusions reached in this work are described in the following sections

7.1 OBSERVATIONS OF THE OBTAINED RESULTS

With the aim to demonstrate the practical usefulness of small pressure perturbation tests to determine the stability margins of a large BWR core, in this work, in similarity with real tests, small steam line pressure perturbations have been used to disturb the reactor.

The Point 3 of Low Flow Stability Tests of Peach-Bottom Reactor chosen for the actual steady state conditions is a nearly stable point. This point is anyway close to the stability boundary in the Power/Flow map and its axial power profile is not bottom peaked (see figure 6.1).

Nevertheless, in the analyzed cases, some characteristics peculiar of the in-phase instability could be recognized using the coupled codes

RELAP5/PARCS; for instance, frequencies in all the oscillations obtained in the analyses varied from 0.3 and 0.5 Hz, i.e., in the typical frequency range of this kind of instability events.

It was observed that the reactor behaviour can be identified as in-phase, because when the time dependent amplitudes of the several modes were calculated with the modal method, it has been observed that the amplitude $n_0(t)$, of the fundamental mode, is clearly the larges while $n_1(t)$ and $n_2(t)$ are almost negligible; in addition, in all the analyzed cases, the simulated evolution of the signals provided by the different LPRMs during the transients is always practically the same.

However in the case C1 the amplitude of the azimuthal modes are not negligible so an association between an in-phase oscillation and two out-phase oscillations can be recognized. This is evident also in the video clip of the average power evolution and in the comparison between the simulated evolution of the signals supplied by opposite LPRM's.

In Peach Bottom-2 Nuclear Power Plant in-phase instabilities appeared and the results obtained by the calculations are found in agreement with this observation.

Considering the analyzed cases, it can be concluded that the core exhibited a large degree of stability at each condition; the only analysis in which the system reaches unstable conditions is case C1, in which the axial power distribution is modified at the beginning of the transient by control rod withdrawal; this analysis shows that the axial power shape affects the instability: the reactor developed an unstable behaviour only after the control rod movement, i.e., after the axial power profile assumed a bottom peaked shape.

In all the analysis performed, at the beginning of the transient, stable conditions were obtained, very similar to the real state of the reactor in the operational point; despite of the good agreement among the calculated conditions and the measured values, the simulated axial power distributions differed from the reference one: in both the cases the power is underestimated in the lower part of the core and overestimated in the upper part.

These discrepancies might be explained by the fact that the Xenon and Samarium distributions used in the calculations might not match the real distribution in the reactor. In fact, since not all the information necessary to perform the analyses was available, the missing thermalhydraulic and neutronic data were taken from the PB2 Turbine Trip (TT2) Benchmark, assuming that the state of the reactor in the Low Flow Stability Tests was about the same that in the Turbine Trip Test 2.

This choice is justified because only 10 days separate the two cycles of tests; nevertheless, even if the reactor conditions in the Low Flow Stability Test and in the Turbine Trip Test are comparable and it is possible to consider the nuclear cross-section distribution unchanged, the choice of using the data of the Turbine Trip Benchmark is an approximation that can produce inaccuracies in the whole analysis because the Xenon and Samarium distributions certainly are modified.

However, even considering all the mentioned limitations, it is possible to state that the present analysis allowed obtaining realistic and meaningful information of the reactor behaviour at the stability boundary of the Power/Flow Map apart. As a product, it was shown that imposing small pressure perturbation offer an operationally simple and precise technique for determining BWR core stability margins.

7.2 CONSIDERATIONS ON THE ADOPTED CODES

All the cases have been analysed with two different nodalizations in order to compare the results obtained with the two different models; similar solutions have been achieved in case A and B but with some differences. In case C the results are completely different and the real reactor behaviour is certainly better reproduced in the calculations performed with the model in which the core is described with 33 T/H channels than in the tests carried out with the model including a single channel.

This is rather obvious, considering the strong coupling of the thermalhydraulic and the neutronic phenomena as simulated in the code. The coupled RELAP5/PARCS code utilizes an internal integration scheme in which the solution of the system and core thermalhydraulics is obtained by RELAP5 and only the spatial kinetics solution is obtained by PARCS. In this scheme, PARCS utilizes the thermalhydraulics solution data (e.g. moderator temperatures/densities and fuel temperatures) calculated by RELAP5 to incorporate appropriate feedback effects into the cross-sections.

The neutronic node structure is more accurate than the T/H one in both of the adopted thermalhydraulic models; so different T/H nodes belong to a neutronic node. It is clear that the larger the number of thermalhydraulic channels is, the more accurate the obtained solution will be.

For example, a very important parameter for determining instability is the void fraction distribution; using the single channel thermalhydraulic core modeling, the void fraction distribution is considered practically uniform over the whole core and this is relatively far from the real situation.

An original contribution of this thesis is the use of the nuclear cross-sections provided by the RELAP5/PARCS calculations to perform modal analysis with the VALKIN code with results very satisfactorily.

In all the performed calculation cases, a very good agreement was obtained between the solution provided by the VALKIN code and the solution achieved with the RELAP5/PARCS coupled codes, taken as reference; the VALKIN code was capable to add more information on the type of the oscillation.

The quantitative disagreement obtained in the Case C1 probably has numerical origin, but only hypotheses can be advanced on its explication because this is a new procedure that should be studied and tested in more transient situation.

The choice of the RELAP5/PARCS results as reference data is justified because, even in the lack of experimental data to validate them, the reliability of the coupled codes have been tested in a great number of different transient situations. Also this work demonstrates the consistency of the data obtained by them; in fact, with two different thermalhydraulic-neutronic coupled codes results very similar for case A1 were obtained. In order to observe the difference between the results achieved by VALKIN using different numbers of modes or different updating times, several transient calculations have been analysed.

Concerning the influence of the number of modes on the results, the differences between the solutions obtained with different numbers of nodes were negligible; therefore, it is possible to conclude that in the addressed cases (stable), the number of modes has a poor influence on the results. As a consequence, to reduce the CPU time, it is more convenient to use only one mode.

For the case C1 it is impossible to make evaluation of this type because for this transient it has been possible to obtain meaningful results only in the calculation with 1 mode.

Considering the updating process it is also noticeable that the updating strategy improves the accuracy of the obtained solutions. Nevertheless, it should be observed that the differences between the solutions achieved with different updating times are not very important; possibly, this occurs because the cases analyzed are practically stable, so in the majority of the test an updating time equivalent to approximately 20 time step has been sufficient to give satisfactory results.

In the Case C1, the only one in which the reactor shows a developed unstable behaviour, the relevant spatial changes make indispensable the updating process and in order to obtain reliable results it has been necessary update the spatial solution each time step.

Other interesting results have been obtained using a modal decomposition of the LPRM's signals performed for case A1 and C1.

Using information on the stable conditions of the system achieved with the steady state VALKIN calculation 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/PARCS transient calculation) was performed. A good qualitative agreement with the results of RELAP5/PARCS transient calculations was observed.

Then, with a great saving of time, simply considering the signals from one of the axial level of LPRM's (43 LPRM) it is possible to achieve the same qualitative information as is obtained from the detailed nodal analysis, for which it is necessary to know the power distribution in all the reactor nodes (23712 nodes) at each time step.

This result is of great practical importance because demonstrates that, in theory, with this methodology, it is possible in a nuclear plant to make an on-line analysis of reactor stability: with the data on the stable conditions of the reactor it is possible to perform a steady state calculation with the VALKIN code obtaining the values necessary to carried out the on-line modal decomposition; then, analyzing the local power signals, provided by the LPRM, it is possible to watch over continually the potential development of instability behaviour in the system and to study it.

7.3 AREAS OF THE FUTURE INVESTIGATION

As mentioned, in this work for the first time, the nuclear cross-sections provided by the RELAP5/PARCS transient calculations were used to perform signal modal analyses with the VALKIN code, for a reactor in which in-phase instability occur. Further investigations should concern the use of this methodology to the study of out-of phase instabilities.

In the Case C1 the reactor has showed a particular behaviour that differs from the experimental data; in fact it has been recognized both contributions of in-phase instability and also out-of-phase oscillations.

This transient needs a deeper analysis but this is possible only after the resolution of the numerical problems founded.; so the future studies should concern the numerical aspects of this new procedure.

In addition, the very good agreement between the results obtained with the coupled codes RELAP5/PARCS and TRAC-BF1/VALKIN make very interesting a more in depth investigation of the effects on the solutions of the use of different thermalhydraulic–neutronic coupled codes.

Finally, another interesting development could concern the testing of the capabilities of the thermalhydraulic code used. More precisely, observing that all the calculations have been computed using the semi-implicit integration method, an interesting aspect appears the study of the same transients with a different integration method. Specifically, the RELAP5 allows the use of a nearly-implicit advancement scheme, and this option is still under development and assessment; so, an analysis of this aspect type could be very important to improve the knowledge of the behaviour of this thermalhydraulic code.

7.4 PUBLICATIONS

Part of the work proposed in this thesis has resulted in papers presentedat international meetings and the contribution to a Project Report.

F. Maggini, R. Miró, F. D'Auria, G. Verdú, D. Ginestar
"Peach Bottom-2 Low-Flow Stability Tests with RELAP5/PARCS" Sociedad Nuclear Española, 29 Reunion Annual Zaragoza 2003, Spain

F. Maggini, R. Miró, F. D'Auria, G. Verdú, D. Ginestar
"Peach Bottom Cycle 2 Stability Analysis using RELAP5/PARCS" International Conference Nuclear Energy for New Europe 2003, Ljubljana September 2003, Slovenia.

F.Maggini, A.M. Sanchez, R. Miró, G. Verdú, F. D'Auria, D. Ginestar
"Peach Bottom-2 Low-Flow Stability Test using TRAC-BF1/VALKIN and RELAP5-mod 3.3/PARCS codes" fourth Crissue Meeting, Stockholm, June 2003

R. Miró, G. Verdú, A. M. Sánchez, F. Maggini, R. Uddin
"Recent Activities and Findings on BWR Stability Analysis: Peach Bottom Low Flow Stability Test, PT3" Last CRISSUE_S Meeting, Pisa, 11-12 December 2003

Contribute to a chapter of the book:
CRISSUE-S-WP-2 (PART 2 OF REAC-SOR) NEUTRONIC/THERMAL-HYDRAULICS COUPLING IN LWR TECHNOLOGY: STATE-OF-THE-ART REPORT DIMNP NT 520(3) Pisa, December 2003

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