Chapter 1

Introduction

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1.1 Motivation

In 2000, 438 Nuclear Power Plants (NPPs) produced 2’468 TWh of electric power, covering 16% of the worldwide and 42% of the West European electricity consumption. Even though many different reactor designs exist, 80% of all reactors are of two basic types. The pressurized water reactors (PWRs) are leading in numbers with 258 operating plants, followed by the boiling water reactors (BWRs) with 90 plants in operation. Naturally, both designs have their advantages and disadvantages.

The local power generation in the core of a nuclear reactor is directly related to the neutron flux, which itself is a function of the reactivity. In BWRs, the reactivity depends strongly on the core void fraction. Thus when a void fraction oscillation is established in a BWR, the power oscillates according to the neutronic feedback. This feedback mechanism which is shown in Figure 1.1 in a simplified manner (see also Figure 3.1 on page 24 in Chapter 3 for full detail) may under certain conditions lead to poorly damped or even limit-cycle power oscillations. Their frequency lies around 0.5 Hz (about twice the transport time of the coolant through the core). Amplitudes from nearly 0% to more than 100% in power have been observed. The oscillations are mostly global, i.e. “in-phase”.

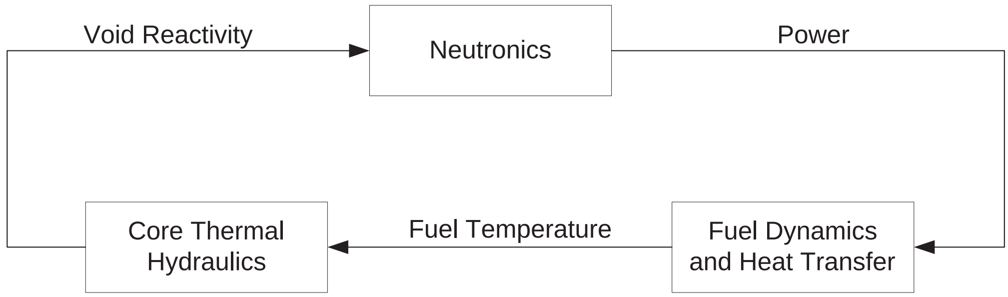

Figure 1.1: Simplified Neutronic/Thermal-Hydraulic Feedback System in a BWR

Higher mode power oscillations are also possible; these divide the core into two regions oscillating in opposite directions at constant overall power. These regional oscillations are cumbersome for the operators since their detection is not directly possible with standard instruments that display only core-average data. Even more complex modes of instability have also been observed.

In addition to the regional oscillations, also local oscillations are possible. Under certain conditions it is possible, that one or a few fuel channels are oscillating independent of the core. The two phase flow inside a channel allows density wave oscillations without the presence of a neutronic feedback ([8] and many others). This effect may dominate the neutronic/thermal-hydraulic feedback loop in rare occasions. For example, in 1997 a badly seated fuel assembly caused local oscillations in Forsmark [4].

During the early years of BWR technology, there was considerable concern about the possible effect of coupled neutronic/thermal-hydraulic instabilities. However, after various in depth experiments and analyses, it became clear that BWRs could be designed such that instabilities would not occur under normal operating conditions.

In addition, the BWR stability issue is no major industry safety problem from a technical point of view. Given appropriate instrumentation, power oscillations are easy to detect and there exist simple, as well as effective, counter measures. A scram will normally solve the problem, even though other less drastic measures will normally suffice. Furthermore, normal operating points in power and core flow tend to be very stable. Stability problems may only arise during start up or during transients which significantly shift the operating point towards low core flow and high power. Therefore, the operating instructions for BWRs contain clear rules on how to avoid operating points (regions) that may produce power-void oscillations. Figure 1.2 shows a power-flow map for the Leibstadt NPP. The lower right side of the plot marks the allowed operating region, the gray regime may only be entered if special measures are taken and finally, the black regime is forbidden due to stability concerns.

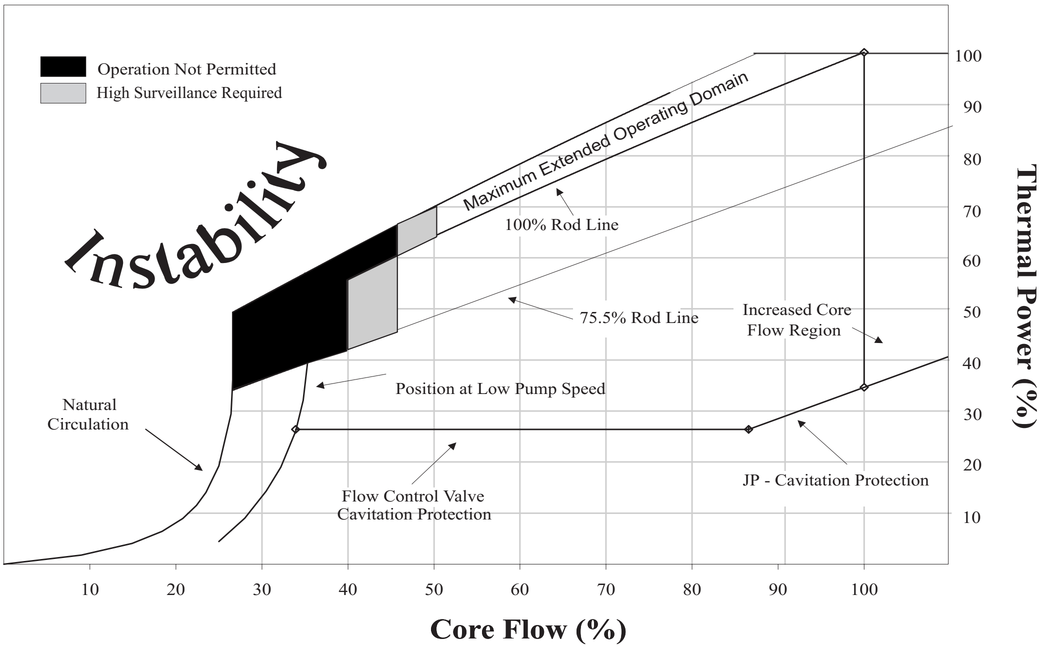

Figure 1.2: Instability Region in the Power-Flow Map

The numerous modifications in “reactor size”, reactor power, fuel design, power density, discharge burnup and loading strategies changed the core stability behavior of the BWR reactor to a significant extent. In comparison to the situation in the seventies, the region of the power-flow map which has to be avoided grew to a respectable size. Several plants, e.g. Caorso and La Salle (see Table 1.1), accidentally entered the less stable region and actually experienced power-void oscillations.

Since, for economic reasons, the trend towards smaller-diameter fuel rods and different loading strategies will persist, the stability problem has to be taken care of. It is, therefore, important to understand the underlying mechanism of power-void instability as thoroughly as possible, as well as to be able to detect and predict power oscillations with the aim to sup-

Date	Plant	Country,	Event

overall good agreement between the computational and mathematical analyses, and the experimental tests and the actual operational BWR incidents, has led to a better understanding of the instability phenomenon.

1.2.1 Selected Papers

The broad literature on BWR-stability covers a wide range of subjects. Reviews of most aspects of nuclear coupled stability may be found in [22], [57], [67].

Apart from some very interesting but limited approaches addressing the issue from an analytical point of view, e.g. the analysis of March-Leuba [54], most investigations involve measurements and/or extensive numerical simulations. Hence, a wide variety of codes (see Table 1.2) have been developed to predict and/or analyze the BWR-stability phenomena.

March-Leuba et al. [54] developed a phenomenological model to simulate the ”qualitative” behavior of BWRs. They also developed a detailed nodalized (numerically discretized) physical model to simulate the dynamic behavior of the Vermont Yankee BWR over a broad operating range by varying the power and flow [55]. Their analysis led to the conclusion that for a wide range of oscillation amplitudes, no significant effect on the integrity of the fuel is expected. However, in order to keep their model simple, they assumed that the coolant enters the core at saturation enthalpy (not as sub-cooled liquid), i.e. that the boiling boundary always is at the bottom of the core channels.

Belblidia et al. [5] adopted a nodal approach to describe the neutron kinetics of a BWR core. They subdivide the core into radial zones and develop nodal kinetics equations by introducing inter-nodal coupling. Using this model they showed that the point kinetics model representation of BWR neutron kinetics yields conservative results, but that for better assessment of BWR core stability, radial coupling effects should be included. However, Karve [39] showed, that the point kinetics model representation may yield dramatically nonconservative results by failing to capture the important effect of the first harmonic mode, which may lead to regional power oscillations in a BWR. Only a three-dimensional neutronic model or a modal kinetic representation will predict those oscillations.

Peng et al. [78] developed a linear frequency-domain computer code NUFREQ-NP for BWR stability analysis under conditions of either forced or natural circulation. That code is based on a one-dimensional drift flux model for the two-phase flow, a simplified 3-D neutronic model and takes into account sub-cooled boiling, arbitrary nonuniform axial and radial power shapes, distributed local losses, detailed fuel assembly dynamics, and system pressure perturbation. The results were compared with experimental data from Peach Bottom-2 stability tests and showed good agreement.

Bergdahl et al. [7] conducted a series of noise measurements at Forsmark-1 and 2 to investigate the oscillations at low-flow/high-power operating conditions during reactor start-up. The LPRM signals in the measurements on Forsmark-1 indicated that these oscillations, which occurred at 0.5 Hz, were generally in phase throughout the core. However, the oscillations varied in strength in different radial positions of the core. Furthermore, the upper

LPRM signals were influenced by the lower ones in the same probe due to void transport upwards in the core. They also noted that the decay ratio (DR) ranged above 0.7, instead of the value of 0.6 predicted by a large computer code.

Valtonen [106] validated the RAMONA-3B three-dimensional BWR transient analysis code, and the TRAB one-dimensional BWR transient analysis code, using data from an oscillation incident that occurred at the TVO I BWR. It was shown, that both regional and global oscillations are possible in BWRs operating in the low-flow/high-power region of the power-flow map. Sensitivities to the inlet orifice of the core, the fuel gap conductance, the axial power distribution and the fuel type were studied. It was shown that decreasing the fuel gas gap conductance, which leads to increasing the time constant of the fuel heat transfer, has a destabilizing effect.

Rizwan-Uddin and Dorning [102] studied the effects of unheated riser sections that are added to enhance natural circulation in the next generation of BWRs. They found that, for a fixed flow rate, the addition of the riser sections makes the system less stable. They also showed that the feedback recirculation loop plays an important role in reactor stability, and if omitted from the model, can lead to nonconservative conclusions.

Wulff et al. [115] simulated the instability that occurred at the LaSalle-2 power plant (and several other BWR transients), using the Brookhaven National Laboratory Engineering Plant Analyzer (EPA) in order to determine the causes that led to the observed magnitudes of power, flow and temperature oscillations. They found it to be a powerful tool for scoping calculations and for supporting accident management. Although very valuable in many contexts, production codes such as EPA are not very useful for thorough stability analyses or extensive parameter studies in general, because of their complexity and long computer running times.

To study the mechanisms for regional instability, March-Leuba and Blakeman [57] developed a model by modifying the LAPUR code. They studied the effect of the first harmonic neutron kinetics mode on the regional instability. They showed that it has a very important influence on this instability and that because of its effect, there is a region in the power-flow map where an regional instability mode is likely, even though the core-wide mode is stable.

Munoz-Cobo et al. [65] extended the phenomenological model developed by March-Leuba [57] to study both global and regional instability. They replaced the point kinetics equations in [57] by modal kinetics equations which they developed based on λ-modes. Using this model, they showed that global oscillations only appear when the first harmonic mode does not have enough thermal-hydraulic feedback to overcome the eigenvalue separation. Further, they showed that self-sustained regional oscillations could arise due to the different thermal-hydraulic properties of the reactor planar halves, if the modal reactivities have appropriate feedback gains.

Van der Hagen et al. [112] developed methods for obtaining the stability characteristics of global and regional oscillations separately from neutron noise signals. The methods were applied for the Ringhals 1 measurements in 1990, where the decay ratio of the regional oscillation was higher than the decay ratio of the global oscillation but could not be seen, because the signal amplitude of the global mode was much higher than the signal amplitude of the local mode.

1.2.2 Different Types of Computer Codes

Table 1.2 lists the common computer codes for stability calculations available in 1997 [22]. In more recent years, some of the codes were further developed. For example, RAMONA-5 was released, which runs significantly faster than RAMONA-3B and contains better thermal-dynamic models. Despite all improvements in several codes, no code similar to MATSTAB has been developed. The combination of frequency domain, three dimensional neutronics and eigeno(7.1(lue/e.1(igen)4U)-r)-1.cto(1.)-9ila3erois1(o)-39265i1(all6-29(unifreode.)]TJ /F11 1 TJ 1

Table 1.2: Common Stability Codes in 1998

Common codes operating in the time domain						
	Thermal-Hydraulics			**Neutronics**	**Owner /**	**Reference**
Name	**Channels**	**Model**	**eq.**	**Dimension**	**User**	
RAMONA-3B	all	D-F NE	4	3	ScandPower A/S	[114]
RELAP5	a few	2 fluid	6	point kin.	INEL, NRC	[15]
RETRAN-3D	4	slip EQ	5	1	EPRI	[76]
TRACG	a few	2 fluid	6	3	GE	[98]

Common codes in the frequency domain						
	Thermal-Hydraulics			**Neutronics**	**Owner /**	**Reference**
Name	**Channels**	**Model**	**eq.**	**Dimension**	**User**	
NUFREQ NP	a few	D-F EQ	3-4	3 simplified	RPI (USA)	[79]
LAPUR5	1-7	slip EQ		point kin.	ORNL/NRC	[70]
STAIF	10	D-F NE	5	1	SIEMENS	[119]
FABLE	24	slip EQ		point kin.	GE	[22]
ODYSY	a few	D-F NE	5	1	GE	[22]
MATSTAB	All	D-F NE	4	3	FORSMARK AB	[32]

eq.	number of conservation equations for thermal hydraulics
D-F	drift-flux description of mixture
NE	thermal non-equilibrium between phases
Slip	non-equal gas and liquid velocities
EQ	thermal equilibrium between phases

in speed and the fact that FD results translate easily into stability criteria offer overriding advantages, particularly, if only the onset of instability, and not its development in time, is of interest.

1.3 The Scope of this Work

The advances in fuel design which raised the stability questions again, also posed new challenges to the existing codes and the detail of the models involved. From the point of view of a core physicist, it is necessary to have a detailed model, a fast code and results that are both easy to visualize and to interpret. The TD codes satisfy the need for accuracy and a high level of detail, but they lack in speed and produce results which do not relate directly to stability-relevant information. By contrast, FD codes produce quick results which are easy to use but do not necessarily reach the required level of detail and accuracy.

With these problems in mind, Dr. Thomas Smed and his co-worker Pär Lansåker at the Forsmark site in Sweden started the MATSTAB (MATrix STABility) project. Recognizing the potential of this work, Dr. H.-U. Zwicky and C. G. Wiktor from the Leibstadt NPP in Switzerland financed this Ph. D. thesis to support the ongoing efforts and to build up some more expertise at the Leibstadt site. The development of MATSTAB provided not only an in depth knowledge of modeling a BWR reactor, but also the opportunity to gain a deeper understanding of the physical phenomena taking place inside the reactor.

The primary objectives for developing MATSTAB and using a new method for calculating and predicting the onset of instability in BWRs were to achieve short execution times (without loss of prediction accuracy) and to provide the capability of predicting core-wide as well as regional oscillations.

Short execution times are achieved by evaluating the core and system dynamics in the frequency domain. This implies that only the *small* deviations around steady state operating conditions are considered, for which the dynamic behavior of the power-void-feedback mechanism is sufficiently linear to be correctly described by a system of linear equations. This approach is widely accepted for studies that are restricted to operating points below the stability limit where the amplitude of the oscillations is limited. One also observes, that when determining stability criteria from noise measurements of power reactors, linear concepts give the best results (e.g. ARMA models, see section 6.1.1 and [47]).

Accuracy is achieved by the detail of the model. The models used in the well known and widely accepted transient code RAMONA-3B were linearized and adapted to the frequency domain for this work.

The ability to detect *global as well as regional oscillations* was achieved with a less common FD approach. No Laplace transformation of the equations are applied. The resulting large sparse system matrix is addressed directly to profit from the information (e.g. global/regional) contained in the eigenvectors. It is the nature of this FD approach that leads to new ways of visualizing and analyzing the BWR stability and constitutes the novel aspects of this work.

In conclusion it was possible

- to “linearize Ramona”
- to predict successfully decay ratios for Forsmark, Leibstadt and Oskarshamn
- to predict regional oscillations for Leibstadt
- to show the influence of different components of the reactor system on stability

To be able to create a tool like MATSTAB, it was necessary to combine expertise from very different fields such as thermal-hydraulics, neutronics, control theory, advanced numerical methods, sparse matrix techniques and, last but not least, some good programming.

The Norwegian based company Studsvik Scandpower contributed the source code of RAMONA-3B to support the cooperation between the two power plants and the Nuclear Engineering Laboratory (LKT) of the Swiss Federal Institute of Technology (ETH).

Finally, it remains to be said that MATSTAB is programmed in the script language of the high performance computing platform MATLAB (MATrix LABoratory) [59] and is called MATSTAB (MATrix STABility). The code is owned by the Forsmark Kraftgrupp AB, which is entirely open minded with respect to new research programs.