The Abdus Salam
International Centre for Theoretical Physics

United Nations
Educational, Scientific and
Cultural Organization

IAEA
International Atomic Energy Agency

2152-9

Joint ICTP-IAEA Course on Natural Circulation Phenomena and Passive Safety Systems in Advanced Water Cooled Reactors

17 - 21 May 2010

THE BOILING WATER REACTOR STABILITY - UPDATED

F. D'Auria, A. Lombardi-Costa and A. Bousbia-Salah

University of Pisa
DIMNP
Italy

THE BOILING WATER REACTOR STABILITY - UPDATED

F. D'Auria, A. Lombardi-Costa, A. Bousbia-Salah

Dipartimento di Ingegneria Meccanica, Nucleare e della Produzione (DIMNP)
Università di Pisa, Italy
dauria@ing.unipi.it, antonella.lombardi@ing.unipi.it, b.salah@ing.unipi.it

KEY WORDS

Stability, BWR, Coupled Code, Best-Estimate

LECTURE OBJECTIVES

The purpose of this paper is to supply general information about Boiling Water Reactor (BWR) stability. The main concerned topics are: phenomenological aspects, experimental database, modelling features and capabilities, numerical models, three-dimensional modelling, BWR system performance during stability, stability monitoring and licensing aspects.

1. INTRODUCTION

Currently, there are more than 80 BWR operating all over the world that provide an important contribution to the overall electricity supply from the nuclear source. The water boiled in the core produces steam that is directly used to rotate the turbine(s), thus constituting the simplest industrial system built so far that is suitable to transform fission power into electrical power. However, fundamental aspects concerned with the interaction between system thermal-hydraulics and neutron kinetics obstructed or delayed (or made less effective), so far, the exploitation of this type of reactor concept. The BWR stability issue [1], caused in the past a number of concerns and requested specific attention by technologists and researchers.

Although events related to unstable behavior occurred every now and then, stability was not a major issue for many years. Stability problems "may only arise" during start up or during transients which significantly shift the operating instructions for BWR. The same instructions contain rules on how to avoid operating points (regions) that may produce power-void oscillations. Figure 1 shows an example of power-flow map for the Leibstadt NPP (Nuclear Power Plant) [2]. 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.

The current trend of increasing reactor powers and of applying natural circulation core cooling, however, has major consequences for the stability of new BWR designs. These modifications have allowed BWR to work at high nominal power, but they have also favoured an increase in the reactivity feedback and a decrease in the response time, resulting in a lower stability margin when the reactor is operated at low mass flow and high nominal power. Also, the increase in core size has led to a weaker spatial coupling of neutronic processes, which result in a stronger susceptibility to so-called out-of-phase oscillations. 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. Instability events had been observed in commercial plants. Table 1 presents instability cases occurred (mostly) during the twenty years before the issuance of ref. [1] in 1997. Some of the several instability events in BWR plants were inadvertent and other ones were induced intentionally as experiments. These instabilities were identified as periodic oscillations of the neutron flux via instrumentation readings. Essentially, neutronic power signals from Local Power Range Monitors (LPRM) and Average Power Range Monitors (APRM) have been used to detect and study the power oscillations.

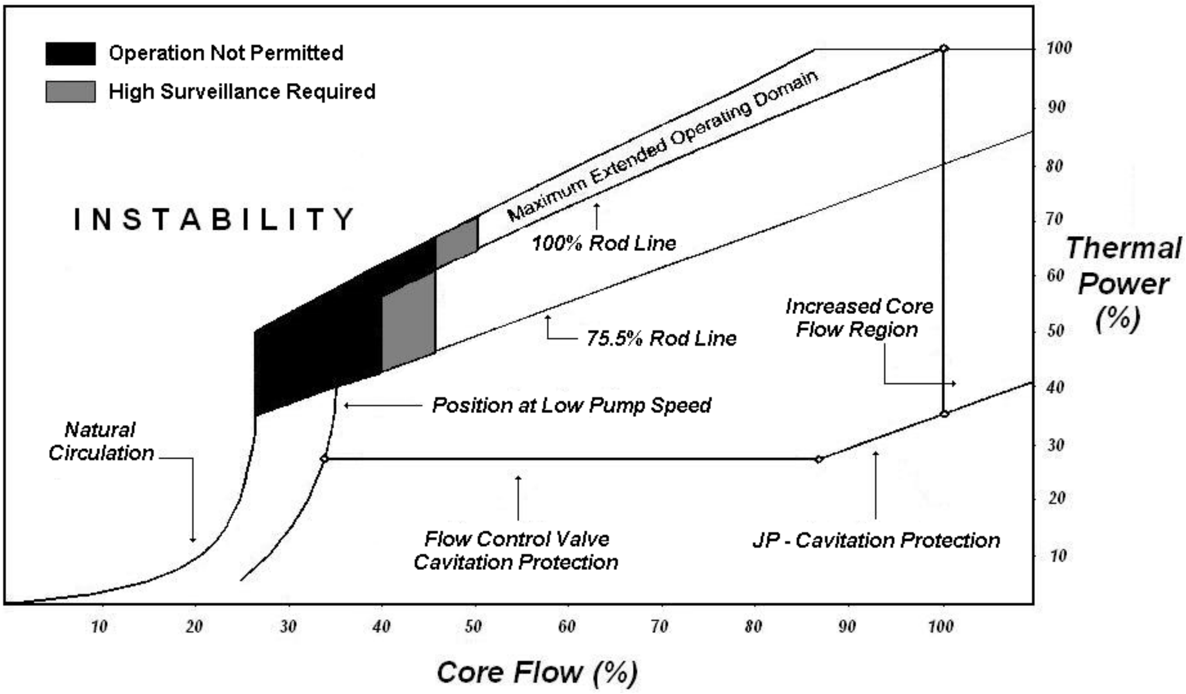

FIG. 1. Instability region in the power-flow map for the Leibstadt NPP.

After the first instability events, authorities in all countries required a review of the stability features of their BWR. The authority requests included specific analyses and changes in the procedures and plant safety systems. The major safety concern associated with instability is the cooling of the fuel and cladding integrity. Consequently, the main objectives of BWR stability analyses could be summarized as follows:

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

The purpose of this paper is to supply general information about BWR instability. The main concerned topics include: phenomenological aspects, experimental database, modeling features and capabilities of computational tools, numerical models, three-dimensional modeling, BWR system performance during stability, stability monitoring and licensing aspects. Further information can be found with more details in the State of the Art Report on Boiling Water Reactor Stability (SOAR on BWR) [1].

2. PHENOMENOLOGICAL ASPECTS

Oscillations in two-phase systems may be originated or connected with different reasons ranging from delays between pressure and density waves propagation velocities, to change in the flow regime, to the interaction between conduction and convection heat transfer, to the feedback between thermal-hydraulic and neutron kinetics parameters, to the presence of different parallel channels and of loops in parallel or in series with boiling channel.

TABLE 1. SUMMARY OF REPORTED BWR STABILITY EVENTS.

Date	NPP, Country, Manufacturer	Event
30.06.1982	Caorso, Italy, GE (General Electric)	It occurred during startup of the reactor. The reactor thermal power was 53.5% and the recirculation flow 38%. The operating point of the reactor entered the unstable region, the oscillations diverged and the reactor scrammed on high-high APRM signal (120% power).
13.01.1984	Caorso, Italy, GE	The second event occurred after trip of a recirculation pump and following loss of some preheater trains. The cold feed water and the strongly peaked axial power distribution led to neutron flux oscillations.
23.02.1987	TVO-I, Finland, ABB Atom	A power oscillation event took place when the plant was brought back to power after a short shutdown period. The reactor power was about 60% with a recirculation flow of 30% when increased APRM signal oscillations started to appear.
09.03.1988	LaSalle 2, USA, GE	Underwent a dual recirculation pump trip following which the unit experienced excessive neutron flux oscillations while it was in natural circulation. The resulting perturbation on the switches for anticipated transient without scram resulted in a trip of both recirculation pumps.
15.01.1989	Forsmark 1, Sweden, ABB Atom	After having carried out tests in several of the selected operating points, the stability boundary was very close. Moving from one point to another, the operator chose to switch from the pump speed control mode to the power control mode. This was done at 71% power and 4700 kg/s. A remaining control mismatch led to a small decrease in core flow. The combined effect of the power control system, caused power oscillations with an amplitude that increased to about ±20% after 20 s, enough the initiate a pump run-down.
26.10.1989	Ringhals 1, Sweden, ABB Atom	In 1989 these NPP was starting up for a new cycle, power oscillations were observed in the core. During the oscillation, several LPRMs gave a high level alarm, indicating that the local power in those positions had exceeded 118 %. The core was stabilized about 30 seconds after the partial scram.
08.01.1990	Oskarshamn 2, Sweden, ABB Atom	A planned power reduction from 106% to 65% was performed. The power was reduced by reducing pump speed to minimum. After about 1 minute, power oscillations had developed, having peak-to-peak amplitudes of ±10%. The APRM signals showed the oscillations to be in phase. A manual partial scram was performed which completely quenched the oscillations.
29.01.1991	Cofrentes, Spain, GE	Oscillations appeared after the operator withdraw control rods and reduced core flow in order to transfer the recirculation pumps from low to high speed.
03.07.1991	Isar 1, Germany, Siemens	The Isar 1 NPP had a trip of four internal recirculation pumps, due to a reduction in seal water flow to the 8 recirculation pumps. The power reduction due to the control rod insertion proved to be too slow to prevent the reactor from entering the unstable region of the power-flow map. Neutron flux oscillations with increasing amplitude appeared at about 50 % power and 30% core flow. When the oscillations reached peak-to-peak amplitude of 30 %, corresponding to a peak power of 67.5 % at 30 % flow, the reactor scrammed on the undelayed setpoint of neutron flux to core flow ratio. This occurred 44 seconds after the trip of the four recirculation pumps.
15.08.1992	WNP 2, USA, GE	The WNP 2 experienced power oscillations during startup. The event occurred early in cycle 8 operation. Upon recognizing the power oscillations, the operators manually initiated a reactor scram. Post event review indicated that the oscillations were in-phase and had grown to peak-to-peak amplitude of about 25 % of rated power.
09.07.1993	Perry, USA, GE	Entry into a region of core instability.
01.1995	Laguna Verde 1, Mexico, GE	During startup of the reactor, at 34% power, the operator was waiting to increase pump recirculation speed. Power was increased to 37% by control rod withdrawal and closure of the control valves of the pumps was initiated, thus leading to power reduction. During these operations the operator observed power oscillations with a tendency to diverge and he stopped the closure of the valves. At this time the peak-to-peak amplitude in the oscillations was 6%.
17.07.1996	Forsmark 1, Sweden, ABB Atom	Local oscillations due to a bad seated fuel assembly.
08.02.1998	Oskarshamn 3, Sweden, ABB Atom	Power oscillations due to a bad combination of core design and control-rod pattern during start up.
25.02.1999	Oskarshamn 2, Sweden,	Power oscillations after a turbine trip with pump runback.

A number of modalities of oscillations, i.e. static, dynamic, with various frequencies, with constant or growing amplitudes, can be identified. The different causes and modalities of oscillations may co-exist and are at the origin of the complexity of some instability occurrences.

2.1 Classification

As a result some classifications of instabilities have been identified. These classifications can be based upon a) fundamental mechanisms, b) pattern of core oscillations, c) system interactions [1].

a) Classification based upon thermal-hydraulic fundamental mechanisms

The static instabilities are:

- Flow Excursion (Ledinegg type);
- Boiling Crisis;
- Relaxation types, including flow pattern transition, geysering and chugging.

The dynamic instabilities are:

- Density Wave oscillations;
- Pressure Drop oscillations;
- Acoustic oscillations;
- Thermal oscillations.

b) Classification based upon geometrical features of the oscillations

In this case the relationships between phases and amplitudes of the waves at different axial and azimuthal positions in the core of a reactor or of a reactor simulator are considered. Coupled neutronic-thermal-hydraulic and pure thermal-hydraulic oscillations can be distinguished, respectively. The following classification results:

- Core wide or in phase oscillations;
- regional or out-of-phase oscillations;
- single channel or local oscillations;
- punctual or not propagating oscillations (their existence can only be envisaged from the performed activity and considered documentation).

Most of the phenomena above can be explained making reference the concept of "harmonic modes" of the reactor diffusion equations.

c) Classification based upon system design

This classification is valid primarily for coupled nuclear-thermal-hydraulic systems (BWR plants), but may occur also in pure thermal-hydraulic systems. The overall system performance and the origin of oscillations are concerned. The following classification can be introduced:

- oscillations involving core-bypass;
- oscillations involving vessel downcomer;
- oscillations involving the entire primary loop (also referred as loop type oscillations);
- oscillations induced by the Balance of Plant, including the control systems.

The outline above demonstrates how the instability phenomena constitute a multidisciplinary subject, involving different technological areas like transient thermal-hydraulics, neutronics, fuel behavior and fuel cycle management, instrumentation, plant control and monitoring and detailed knowledge of plant features and operating conditions.

2.2 Nomenclature

It seems worthwhile to introduce hereafter some definitions that are common to the stability analyses in fluid-dynamics, see also ref. [24].

Acoustic instability: This occurs when standing waves are excited in a single or two-phase system with a frequency in the acoustic range: steam line resonance and acoustic instabilities in the steam dome and upper plenum regions of BWRs have been observed.

Balance of Plant (BoP): The Balance of Plant (BoP) in this report refers to components and systems, inside and outside the nuclear island, necessary to transform the thermal energy into electrical energy with optimized overall efficiency and by hardware and software to control the entire plant performance.

Chimney: The chimney in a boiling water loop is intended as the zone of the riser comprised between the top of the core and the separators region.

Core Boiling Boundary Fraction: This is the ratio between the core power fraction up to the defined core average boiling boundary limit and the core power fraction required for coolant saturation.

Core or stability monitor: This is a sub-system specifically included in the monitoring system that accounts for core stability; the sub-system may only detect core stability or also take (automatically) effective countermeasures.

Coupled neutronic and thermal-hydraulic instabilities: They are the wide variety of oscillations occurring in BWRs which are the result of the interaction between thermal-hydraulic and neutronic phenomena. Core-wide, out-of-phase and local oscillations belong to this class.

Density wave: A density wave is a perturbation in the density of the fluid mixture, which travels along the heated channel with a characteristic speed depending on local conditions.

Dual oscillations: Simultaneous occurrence of in phase and out-of-phase oscillations.

Dynamic instabilities: These terms characterize the wider class of instabilities that can be studied only through the use of time-dependent balance equations.

Exclusion zone: In the BWR operating power/flow map, the exclusion zone is a region not allowed during the nominal operation of the reactor (see Fig. 1).

Flow biased APRM scram: The APRM signal is considered together with the core flow signal before taking control action in BWR plant.

Flow excursion (Ledinegg) instability: This is a type of static instability that is determined by the relationship between the pressure drop characteristic of a boiling channel and the pressure drop characteristic imposed by an external system.

Harmonic modes: Harmonic modes are represented by the eigenfunctions in which the general solution of the partial differential neutron diffusion equation, applied in a given domain (e.g., the reactor core) with appropriate boundary conditions (e.g., zero neutron flux at the extrapolated boundary), can be decomposed.

Kinematic wave: A wave that progresses with the speed of the fluid, either steam or liquid or two-phase mixture.

Monitoring system: Everything at hardware and software level that is used to detect reactor configuration (the core stability monitor is a part of this). For instance, the instrumentation used for measuring neutron flux, coolant flow, core inlet subcooling, pressure and Feed Water (FW) flow is part of the monitoring system.

Neutronic feedback: Thermal-hydraulic instabilities may occur even in out of core loops. Oscillation in thermal-hydraulic quantities may induce oscillations in neutronic related quantities (mostly, neutron flux), in in-core systems. Neutronic feedback identifies the physical variables carrying the (oscillatory) information from the neutron to the thermal-hydraulic domain and the resulting effects. The neutronic feedback occurs mostly (but not exclusively) through the average channel void fraction ("void coefficient") and the average fuel rod temperature ("Doppler coefficient").

Parallel channels: Different fuel elements up to including the entire core are reported as parallel channels. Parallel channel oscillations may be either "core wide" either "regional". These terms (i.e. parallel channel) are mostly appropriate for out of core test loops.

Phase delay: This is the phase shift between the phase of the oscillations of generic signal and the phase of a reference signal.

Pseudo Random Binary Sequence (PRBS): In the field of BWR stability, PRBS is a particular form of a perturbing signal adopted for checking stability in both reactor test conditions and calculations. It consists of alternate high and low values of a given parameter (e.g., a pressure setpoint) which are generally superimposed in a random like fashion to the nominal value with, eventually, some restriction on the mean value. When a PRBS is applied as an external forcing, the related system response can be used to characterize stability.

Single channel: This identifies a single fuel element of a BWR plant. Single unstable channel refers to a situation when only one of several parallel channels is oscillating in a stable system.

Stability boundary: A stability boundary is represented by a relationship between the parameters describing a system status which defines the conditions in which the system shows marginal (or neutral) stability, i.e. in which perturbations are neither amplified nor damped.

Stability margin: A stability margin is a properly defined measure of the distance of a system status from the stability boundary.

Static instability: These terms identify a class of instabilities that can be theoretically explained without the use of time-dependent conservation equations.

Thermal-hydraulic feedback: This includes all the thermal-hydraulic reaction mechanisms in a BWR plant that may intervene once an oscillatory condition arises in the core.

Thermal-hydraulic instabilities: These are identified by periodic time oscillations of various quantities in a boiling system (either single channel either parallel channels). Excursion of heated wall surface temperature may result from thermal-hydraulic instabilities.

THermal Oscillations (THOs): Oscillations involving the heater dynamics in a boiling channel.

3. THE EXPERIMENTAL DATABASE

Loops offer many advantages for studies of two-phase flow thermal-hydraulics. An electrically heated mockup of a fuel assembly in a loop can be exposed to far more onerous operating situations (in terms of channel power and channel flow, for example) than could ever be permitted - for testing purposes - in the core of a nuclear reactor in commercial operation. Therefore, loop studies can be used to prove thermal-hydraulic operating limits of various fuel designs with respect to both dryout and flow stability (including the stabilizing influence of channel inlet orifices of various strengths), and also to study operating situations close to these limits. Moreover, loops can be used to investigate situations which differ significantly from those encountered during normal BWR operation (with regard to pressure, saturation temperature and inlet subcooling, for example). A fuel assembly mockup in a loop can be equipped with far more instrumentation than could be possibly introduced into the core coolant channels of operating BWR. Notwithstanding the relevant importance of experimental data base, there are some drawbacks connected with loop or NPP experiments. For instance they are costly to build and costly to run. This applies in particular to full-scale thermal-hydraulic testing of fuel assemblies at realistic BWR operating conditions with respect to temperature and pressure. Cost reductions can be achieved via the use of scaled-down loops, but problems may then arise when it comes to extrapolating experience gained from such tests to the full-scale geometry. Further cost reductions may be gained by building low-pressure test loops that make use of alternative coolants with reduced boiling points, as compared to water.

3.1 Data from fundamental research

In modern BWR, the number of parallel channels is so large (several hundreds) that the pressure head will remain essentially unaffected, even if one single channel were to experience undamped flow oscillations. On the other hand, when a heated channel in a loop experiment encounters undamped oscillations, the flow through the entire loop will oscillate - which normally influences the pressure drop over the channel. There are two ways of dealing with this problem:

- Introduce a bypass channel; this option demonstrates theoretically that if a test channel in a closed loop experiment is operated with a flow bypass in parallel, then this bypass will exercise a stabilizing influence on the pressure conditions at the inlet and outlet of the test channel - bringing the operating conditions of that channel closer to those representative of actual BWR cores. In this manner, improvements will be gained with respect to the applicability of the test results to realistic situations. However, reaching fully realistic BWR operating conditions in a test loop requires a very substantial bypass.
- Conduct the stability tests at natural circulation; the weight of the coolant in the downcomer of the loop then provides a constant pressure head. In this situation, all the single-phase pressure losses caused by friction in the downcomer and the piping leading to the inlet of the test section must be known accurately, since they all contribute to the flow-stabilizing pressure drop over the channel inlet orifice. Inertial effects in the flow through the same components must also be taken into account.

Unless either of these two measures are taken, flow stability experience obtained from loop experiments will not be directly representative of actual BWR core coolant channel operation. Nevertheless, results from such tests are still of great value for qualifying computer codes developed for numerical analysis of thermal-hydraulic stability. Such benchmarking exercises require that the modeling accounts for all the features of the entire loop which are of importance with respect to the flow dynamics.

3.2 Data from out-of-pile tests

In the late 1950´s and the early 1960´s, the first generation of modern BWR were still in the stages of design and construction. Hence, any thermal-hydraulic issues of interest had to be investigated via either theory or loop experiments. A great number of such studies were made in those days, many of them addressing the topic of hydrodynamic stability of two-phase flows.

Until the Eindhoven Symposium [3], the experimental studies largely addressed flow conditions up to the threshold of instability. Since then, considerable research effort has been directed towards investigations of what happens beyond that threshold. Among this studies, a more recent investigation, in the PIPER-ONE BWR simulator, installed at DCMN of University of Pisa, led to characterize the "punctual" instability, or local instability, where is showed that in some situation, instability does not propagate along the channel axis. Amplitude of oscillations is close to zero at the channel inlet and outlet but it peaks around the zone of peak power [4]. Further experimental out-of-pile tests are described in the Appendix A1 as well as in the SOAR [1].

3.3 Data from in-core experiments

A limited number of experiments finalized to the study of stability, have been conducted in BWR NPPs. Some in-core experiments are described in the Appendix A2 and more details can be found in ref. [1].

4. MODELING FEATURES AND CAPABILITIES

A wide variety of codes and models exists that may be used to address the stability issues, ranging from sophisticate system codes, able to calculate an overall plant behaviour, to very simple models. All of them have the capability to deliver similar results to quantify stability, although reliability may be different. In fact, the objectives in the development and the level of approximation and of qualification, including the reliability of results are different in the various cases. Multipurpose codes solving multi-dimensional equations both for neutronics and thermal-hydraulics are available. On the other hand, simplified codes based on Homogeneous Equilibrium Model (HEM) are still used in the same frame. Furthermore, in some cases qualification for BWR stability applications may include tens of applications to basic experiments, separate effect loop tests and BWR plant occurrences; in other cases HEM based codes can be simply developed and applied.

4.1 Numerical models

Various equation models are adopted for thermal-hydraulics and neutronics corresponding to different physical detail in the description. Space and time discretization of thermal-hydraulic balance equations is made by suitable numerical methods. Finite difference methods constitute a common choice, in which nodes representing the fluid contained in reactor components are connected each other through junctions, describing the related flow paths. The mass and energy balance equations are written in nodes, while momentum equations are written at junctions making use of the staggered mesh technique which is acknowledged to improve numerical stability of the calculation. The donor cell principle is also used for calculating advection terms in mass and energy equations, assigning to the fluid at each junction the properties of the fluid in the upwind region; this principle has the fundamental role of assuring the transportive property of the numerical method, strongly affecting its numerical stability as well.

The type of time discretization adopted for thermal-hydraulic balance equations contributes to determine the effect of truncation errors on the calculated results. In particular, diffusion is responsible for smoothing out sharp propagation fronts in the calculated results, and for damping of free oscillations as a consequence of spurious energy dissipation. The latter phenomenon is particularly relevant for the simulation of thermal-hydraulic instabilities, and may lead to calculate stable conditions, whereas unstable ones would be expected on the basis of physical models.

In addition to the above described techniques, widely adopted in reactor thermal-hydraulics, time domain codes for the analysis of BWR stability adopt methods specially developed in the aim to improve computational efficiency or to minimize truncation error effects. Integral formulations of momentum equations and higher order integration schemes are examples of these techniques.

A complementary effort has focused on the development of reduced order models, often consisting of a limited number of ordinary differential equation that represent the most important dynamical processes of a BWR. Muñoz-Cobo et al., [5], used the DWOS_M_SU (Density Wave Oscillations with Modal Kinetics and Sub-cooled Boiling) code to investigate the Cofrentes instabililiy event occurred in 1991. The paper presents a study about the influence of the sub-cooled boiling on the feedback mechanisms that lead to the development of out-of-phase instabilities in boiling water reactors. In the sub-cooled boiling region the bubbles are formed in the walls (clad surface), and detach from the walls to travel to the sub-cooled region of the channel where they collapse. Because reactivity feedback is very sensitive to void variations and is axially weighted by a distribution factor that depends on the square of the power distribution, the void fraction variations in the sub-cooled boiling region are expected to give a large contribution to the void feedback reactivity.

The power transferred to two parallel channels, possibly simulating the core of a nuclear reactor, versus time is displayed in Fig. 2. As can be observed, the two quantities are out-of-phase, i.e. while in one half of the reactor core the power attains its maximum value at the other half of the reactor core the power attains its minimum value and the total core power is (nearly) a time constant.

Stability and semi-analytical bifurcation analyses have been performed using the bifurcation analysis code BIFDD to determine the stability limits for both in-phase and out-of-phase oscillation modes, as well as the nature of PAH (Poincare´–Andronov–Hopf) bifurcation, ref. [28]. The results show that both sub- and supercritical PAH bifurcations are encountered in different regions of the parameter space. Furthermore, it has been seen also that analyzing the properties of the elements of the eigenvectors gives complete information on the type of oscillation mode (in-phase or out-of-phase). A distinction could be made between in-phase and out-of-phase oscillation modes, on the one hand, and the fundamental and first azimuthal neutron kinetics modes on the other hand.

4.2 System codes capable of simulating the BWR

Computer programs are adopted to evaluate stability of BWRs and other boiling channel systems. In describing them, two classes are considered following a generally adopted classification:

- Frequency Domain Codes, whose purpose is the linear stability analysis of BWRs or other boiling systems. In the frequency domain, perturbing and Laplace transforming the neutron kinetics equations allow to easily include the fission power dynamics into the linear model for BWR stability. Examples of frequency domain codes can be seen in the Appendix B, part B1 (Tables B.1 to B.9).

- Time Domain Codes, which include analysis tools specifically developed to simulate the transient behaviour of plant systems; these codes have the capability to deal with the non-linear features of BWRs and are based in simulation techniques. Examples of time domain codes can be seen in the Appendix B, part B2 (Tables B.10 to B.26).

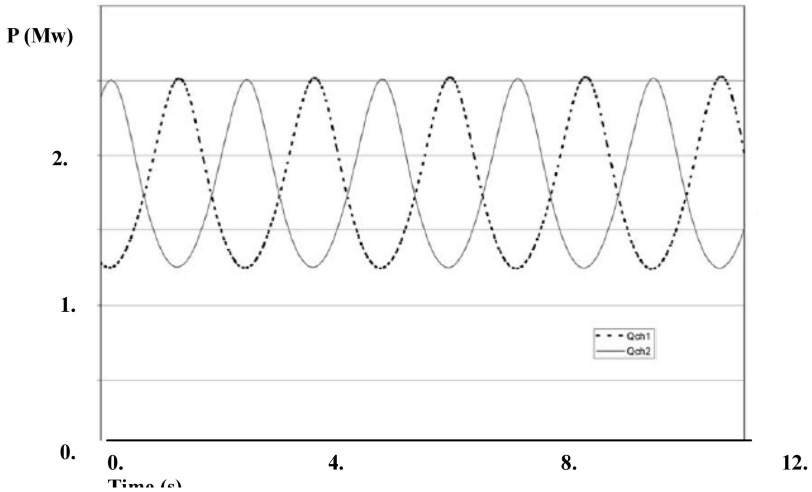

FIG. 2. Power transferred to two parallel channels simulating the core of a nuclear reactor versus time.

4.3 Current capabilities of system codes

Available models cover the whole range of phenomena observed in power reactors and experimental loops. This means that they are able to provide physical understanding of the evolution of meaningful quantities describing the transient behavior of unstable boiling systems. It can be observed that the range of phenomena shown by models includes but it is even wider and denser than the one observed in experiments and reactor occurrences. In particular, the range of phenomena predicted by models is wider in connection with the extreme consequences of nonlinearities, e.g. the quasi-periodic or the non periodic behavior.

The Appendix B presents information about codes whose application is most frequently reported in the literature concerning BWR stability. A more extended description can be found in the SOAR, [1], where information is included in relation to fuel dynamics and heat transfer, to the influence and the modeling of ex-core systems and to code qualification studies relevant to BWR stability.

4.4 Three-dimensional (3D) modeling

Nowadays, the coupled codes method, which consists in incorporating three-dimensional (3D) neutron kinetics modeling of the reactor core into thermal-hydraulic system codes, is extensively used. It is particularly suited for simulating transients that involve core spatial asymmetric phenomena and strong feedback effects between nuclear fission phenomenology and reactor loop thermal-hydraulics. The recent 3D nodal neutron kinetic models usually employ meshes with vertical axis that at the planar level (orthogonal to the axis) have the size of the fuel assemblies (or part of assemblies). In most of events, the complex three-dimensional nature of in-core coupled neutronic-thermal-hydraulic coupling processes will not play an important role in macroscopic plant behaviour. However, events exist in which the 3-D power distribution is strongly dependent on the plant dynamic performance [6]. Application of coupled code methods to the simulation of reactor events also involving stability, are discussed in ref. [25] (additional comments are provided in section 5.2 below).

4.5 Prediction by system codes

The available literature on codes adopted for BWR stability reports many successful applications to out-of-pile and in-pile experiments, showing good agreement between observations and predictions. This supports the conclusion that presently available codes are capable to provide a detailed quantitative explanation of phenomena.

The coupled code RELAP5/PARCS was used by Costa et al. [15], see also ref. [26], to characterize the unstable behaviour of the BWR NPP Peach Bottom unit 2 during the “Low-Flow Stability Test” at the test point 3, ref. [7]. The experiment was performed in the right boundary of the instability region of the power/flow map, i.e., at around 38% core flow rate 59.2% power, as shown in Fig. 3.

The steady average axial power distribution measured and calculated for the Peach Bottom experiment is given in Fig. 4 together with the calculated radial power distribution, ref. [26]. For the perturbation analysis, the outlet reactor pressure was perturbed by a twin-peak signal having amplitude equal to 0.055 MPa. After the perturbation, the pressure oscillation turned to stable trend in few seconds. A corresponding behaviour was observed for the core power which exhibited damped oscillations with typical frequency values ranging between 0.3 and 0.4 Hz as given in Fig. 5.

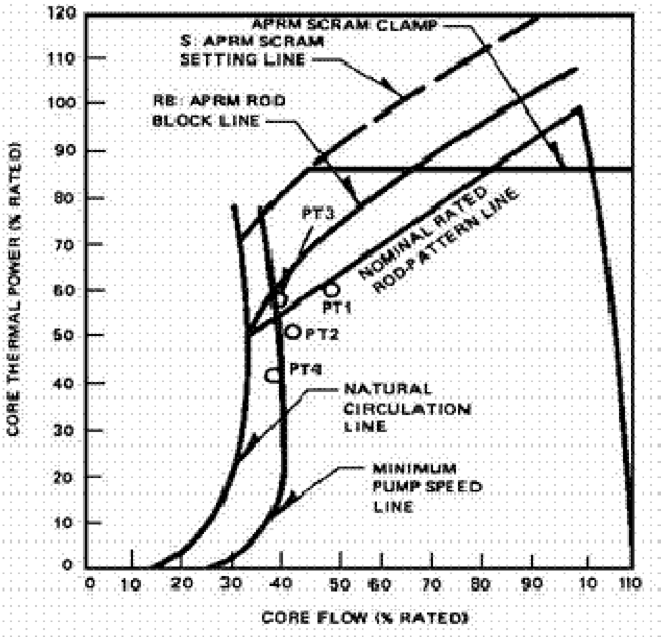

FIG. 3. Peach Bottom-2 low-flow test conditions.

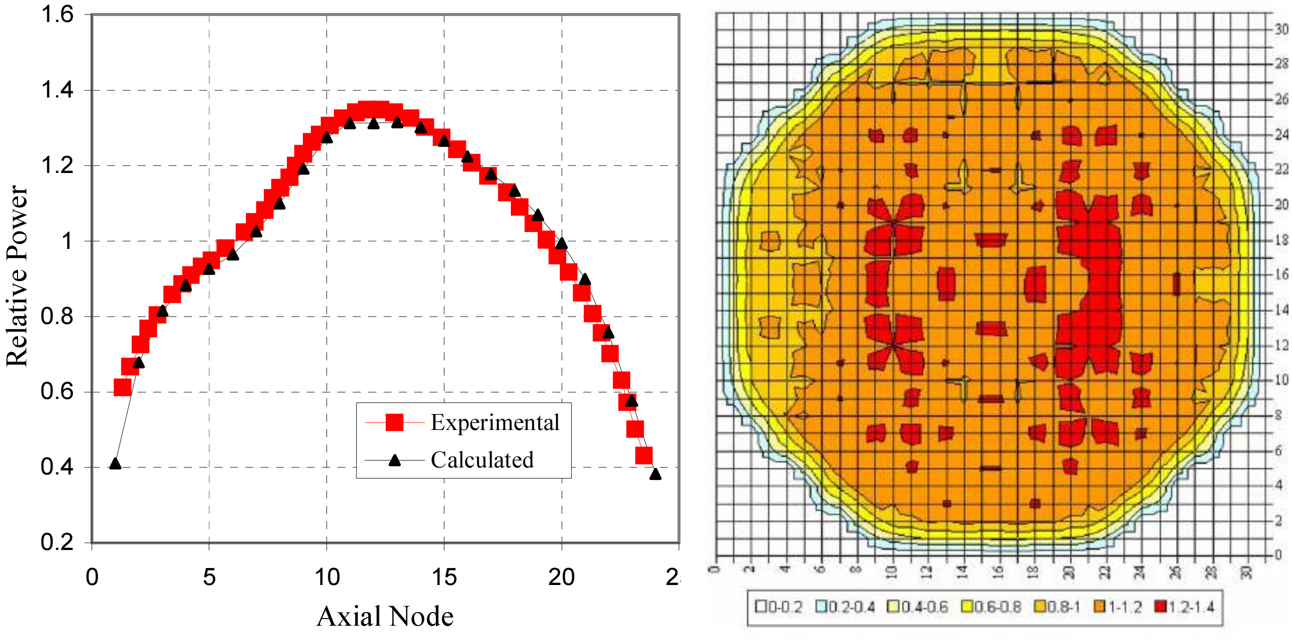

FIG. 4.Peach Bottom-2 steady state: left) average power axial distribution; right) radial power distribution.

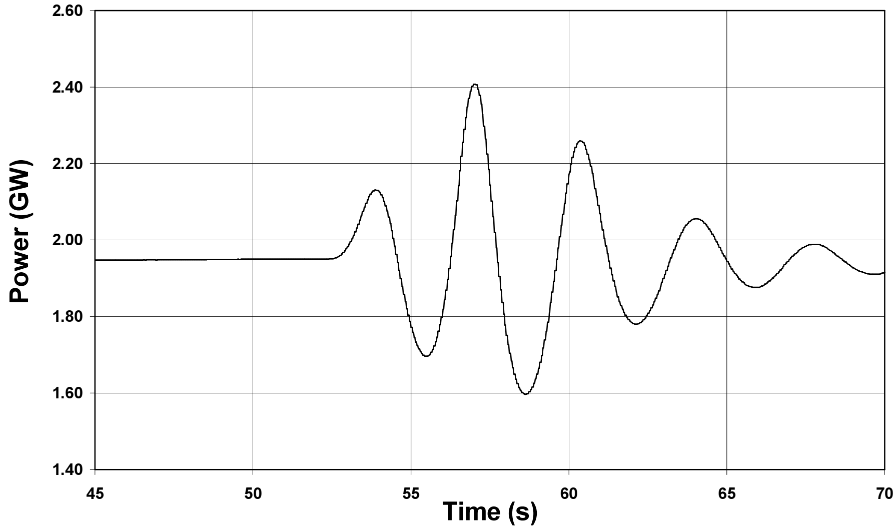

FIG. 5. Peach Bottom 2BWR-stability transient: calculated value of total reactor core power.

Still in relation to the coupled neutron kinetics thermal-hydraulic code, Solís et al. [16] developed a temporal adaptive algorithm to perform the synchronization and optimization of the performance of TRAC-BF1/NEM/COBRA-TF code also involving the "sub-channel" capability. Among the mentioned computational tools, the COBRA (Coolant Boiling in Rod Arrays) series of programs was developed for the detailed analysis of boiling in rod bundles. The COBRA programs use a three-field representation of two-phase flow involving the separate modelling of the liquid film and the droplet fields. This code can only be used to analyze the core region alone and must be coupled to a system code, such as RELAP5-3D or TRAC-BF1, to perform an integrated analysis of the entire system [17].

Within the same framework considered above, fast and reliable methods to integrate the neutron kinetics diffusion equations have been developed and applied. Miró et al. [18] proposed a modal method to integrate those equations in which the spatial part has been previously discretized using a nodal collocation method. The method has been implemented in a code called MODKIN. Five calculation cases are discussed in ref. [18]; one of these was the behaviour of two operational points of the NPP Leibstadt (cycles 10 – in phase oscillation – and 7 – out-of-phase oscillation). The MODKIN and the RAMONA codes were coupled to obtain detailed information regarding the state of the reactor for each integration time step. Namely, the nuclear cross-sections provided by RAMONA allowed the calculation of the nuclear constants for the reflector nodes surrounding the reactor and the whole set of cross section is used as an input for the MODKIN 3D code. The power evolution for the Leibstadt in-phase oscillation case is represented in the Fig. 6. The good agreement among the results calculated with both codes, both predicting a limit cycle, can be observed.

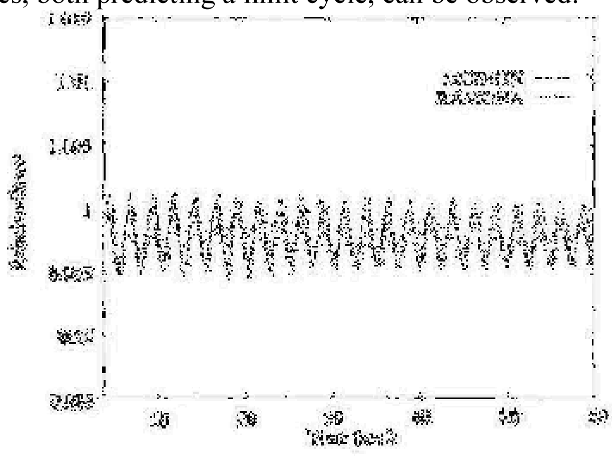

FIG. 6. Power evolution for Leibstadt in-phase case.

Hotta et al. [19] studied the applicability of TRAC/BF1-ENTRÉE to the regional instability. The fidelity of the numerical model was studied with regard to the thermal-hydraulic mechanism (density-wave oscillation) based on the FRIGG-4 loop test results. Electrically heated rods were arranged in three concentric circles. After measuring basic steady-state properties, such as pressure loss and void fraction, the stability limit power was measured at 11 natural circulation flow points by monitoring flow fluctuation with increasing power level. Among other conclusions, fidelity of the code was demonstrated in predicting the stability limit power and limit-cycle amplitude for a wide range of system pressures (based on the FRIGG-4 loop test). After, based on these results, Hotta and Ninokata [20] applied the system to the coupled neutron kinetics - thermal hydraulic instability analysis and demonstrated that the system can predict the regional decay ratio under a wide range of operating conditions.

Hotta et al. [21] also presented simulation results of the coupled codes TRAC/BF1 and ENTRÉE based on the one-pump trip test performed in a Japanese BWR-5 plant. This reactor produces the rated heat output of 3293 MW (thermal), and the core was composed of 764 fuel assemblies. In the one-pump trip test, one of two recirculation pumps was intentionally tripped initiating a rapid decline in the core flow rate. The test was initiated at a relatively high-power (98%) and low-flow (86%) condition. Selected results are given in Figs. 7 (a) and 7 (b). The axial power profiles based on the bypass moderator density feedback were compared with the measured ones at two states. The inclusion of the moderator density feedback model improved the transient prediction since this would lead to less neutron slowing down in the upper region, and thus, the relative power level was suppressed in this part. Looking at the final state, Fig. 7 (b), the measured power skewed slightly upward due to the local control blade insertion and the calculation results slightly over-estimated this trend.

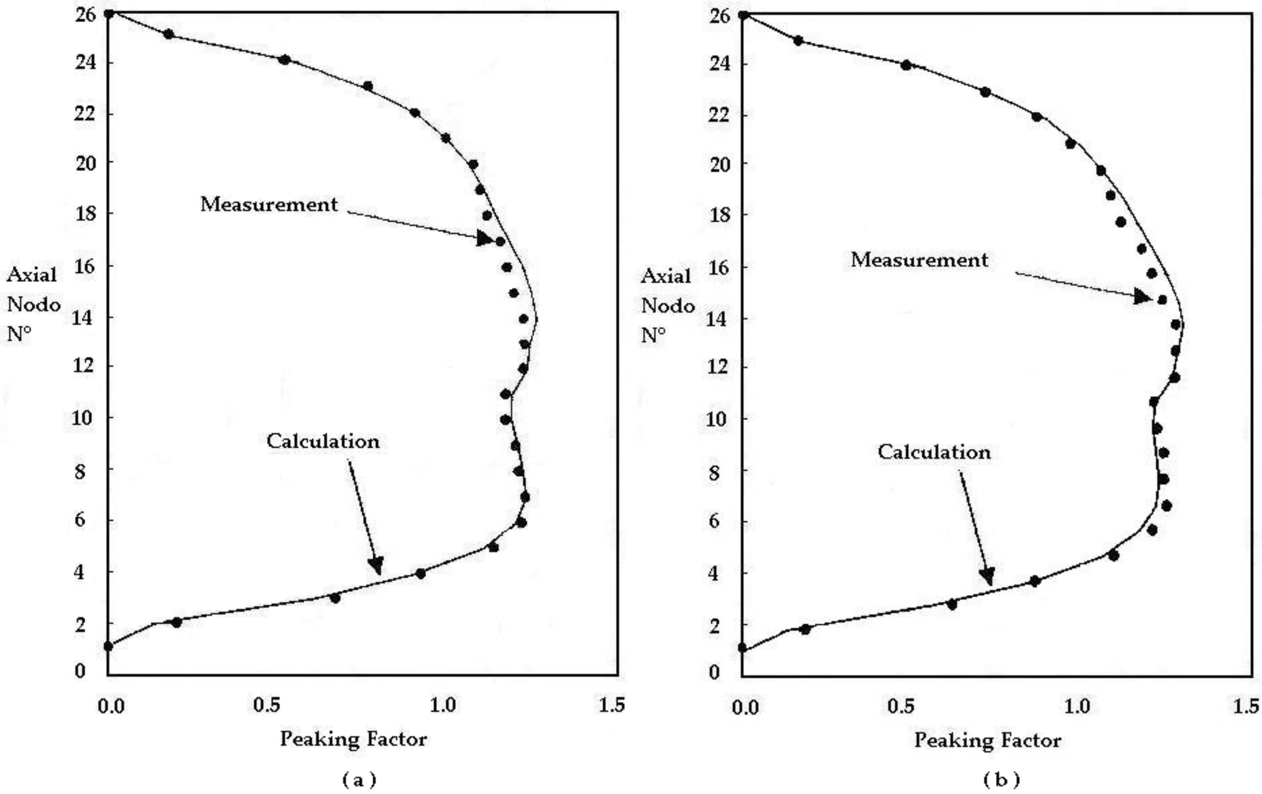

FIG. 7. Japanese BWR-5 NPP: correlation between calculated and measured axial power profiles at (a) initial state (0 s) and (b) final state (80 s).

Lin et al., [27], found that core conditions at end of cycle (EOC) result in larger decay ratios than those at beginning of cycle BOC in the Kuosheng BWR/6 stability analysis performed by the frequency domain code LAPUR-5. Detailed sensitivity analyses performed by the same authors concluded that density reactivity coefficient, delayed-neutron fractions and decay constants, core power axial shape, and total core flow are the most important parameters that might affect the accuracy of decay ratios.

5. THE BWR SYSTEM PERFORMANCE DURING STABILITY

5.1 NPP data (planned transients and un-planned events)

Some of the several instability events in BWR plants were inadvertent and other ones were induced intentionally as experiments. Several tens of stability tests have been performed in BWR plants, ref. [1]. Typical examples are outline below.

Planned transients

<u>Peach Bottom tests</u>
One of the early stability tests was conducted at Peach Bottom-2 in April 1977 [7]. The reactor design was a 3293 MWt GE BWR-4 type and the stability tests were conducted along the low flow end of the rated power flow line, and along the power flow line corresponding to the minimum recirculation pump speed. The reactor core stability margin was determined from an empirical model fitted to the experimentally derived transfer function between core pressure and the APRM signals.

Four test conditions for the low-stability tests were planned to be as close as possible to one of the following reactor operating conditions:

1. points along the rated power-flow control line (PT1 and PT2);
2. points along the natural circulation power-flow control line (PT2, PT3 and PT4);
3. extrapolated rod-block natural circulation power (test point PT3).

The main objective of these tests is to provide a data base for the qualification of transient design methods used for reactor analysis at operating conditions. The tests were performed in the right boundary of the instability region of the Power/Flow (Figure 3). During the stability tests a strong effect of the xenon was observed.

<u>TVO-2 tests</u>
In November 1980, stability tests under natural circulation conditions were integrated into the normal start-up testing at TVO-2 [8]. This is an ABB ATOM internal pump type BWR with 2000 MWt rating. The reactor stability margin was evaluated by Fourier inversion of the transfer function from reactor pressure to APRM. This was obtained from measurements during sinusoidal perturbation of the turbine admission valve positions. Under natural circulation, established by tripping the pumps sequentially, four of the six pumps stopped and remained stationary while two pumps continued to rotate driven by the flow and by inertia and connected with the respective engines. After the pump trip, the reactor power was increased by control rod withdrawal. Self sustained neutron flux oscillations were recorded at 57.2% core power. The APRM signal oscillates almost purely sinusoidally at 0.34 Hz, with an amplitude of 5% of the rated power and in a global mode, i.e. the LPRM oscillated in phase at all radial locations.

<u>Vermont Yankee tests</u>
The stability tests at Vermont Yankee in March 1981 [9], was one of the test during which oscillations were detected. The plant is equipped with a GE BWR-4 type reactor with 1593 MWt rated power. The

tests were performed with recirculation pumps stopped (natural circulation) and with all pumps running at minimum speed. Control rod patterns were adjusted to achieve relatively high power levels. A limit cycle condition was achieved without perturbation at the highest power value under natural circulation. The limit cycle oscillation was suppressed when a few control rods were inserted slightly. Neither unusual behaviour nor equipment damage occurred during the experiments.

KRB-B and KRB-C tests

The twin unit nuclear power plants KRB-B and KRB-C, each having a rated power of 3840 MWt, are BWR designed by KWU-Siemens. KRB-B and KRB-C stability tests were performed in April 1984 and November 1984, respectively [10]. The tests showed that both units were stable in normal operating range, and stable at a power level above the 100% control rod line at minimum recirculation pump speed. The limit cycles observed in neutron flux at slightly above the stability threshold power level, were characterized by out-of-phase oscillations: half of the core appeared to be out of phase with respect to the other half. The amplitude of APRM was only a fraction of the amplitude observed in LPRM. The out-of-phase oscillation mode developed from an initially more or less random situation and settled into a stable mode later on.

KKL (Leibstadt) set of data

NPP stability data for Leibstadt (KKL) NPP were recently made available, ref. [29]. A similar test procedure was employed at the plant: the stability tests were carried out after having operated the core at full power for several days in order to achieve Xe-equilibrium conditions; stability tests were performed at beginning-of-cycles 10, 13 and 19.

A 10-min recording was made after the targeted operation conditions have been reached for any given set of neutron flux and process signals using the same data acquisition system. (A lower sampling frequency was used for the cycle 10 tests.) A total of 42 signals are measured during each test, including the totality of eight APRM signals, eight LPRM signals located at the bottom part of the core and a variety of process signals such as the core flow, steam flow, feed-water flow, etc.

The objective of the cycle 10 core stability tests was to investigate the stability characteristics of a very heterogeneous core (with regards to fuel bundle design and fuel vendor), Tab. 2, at different conditions in the power-flow map (3138 MW at that time), Fig. 8. The cycle 13 tests were carried out with the primary goal to validate the exclusion regions up to the permitted uprated power level of 112% power and to check the performance of the online stability monitoring system that had been installed at the plant. The cycle 19 tests were performed in order to validate the maximum extended operating domain (MEOD) line in the updated operating map as well as the analytically defined boundaries for stability monitoring, i.e. monitoring and excluding regions, concluding the power uprate process of the plant to 3600 MW.

The results from the application of a method proposed by the same authors in ref. [29] to evaluate Decay Ratio DR and Resonance Frequency (RF) are reported in Fig. 9. The KKL core was found to be stable in all the investigated operating points of each cycle, except for one operating point in cycle 10 which yields a DR value close to 1 indicating a possible limit cycle oscillation.

TABLE 2. OVERVIEW OF KKL STABILITY TESTS DURING CYCLES 10, 13 AND 19, ref. [29]

Cycle	Date	Core loading	Stability tests number	Sampling frequency (Hz)
10	September 1993	(1) 50% GE fuel bundles (including 8 × 8, 10 × 10 designs) (2) 50 % SVEA-96 fuel (10 × 10 design)	1–10	10
13	September 1996	100 % SVEA-96 10 × 10 Core	11–18	40
19	September 2002	(1) 40% SVEA-96 10 × 10 (2) 60 % SVEA-96 Optima with PLR	19–28	40

Note: PLR, part length rod.

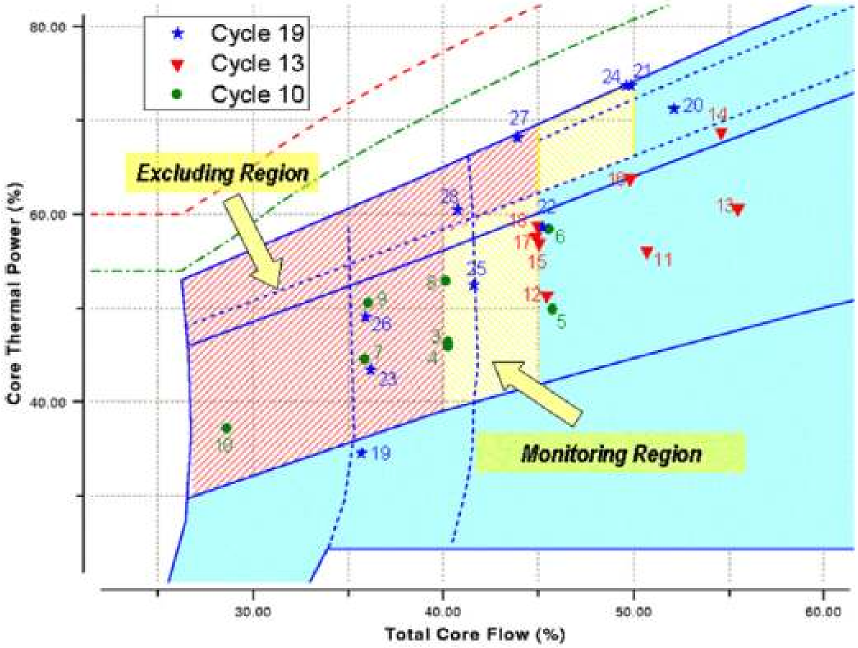

FIG. 8. Power-flow map and stability measurements performed during cycles 10, 13 and 19 in the KKL NPP, ref. [29].

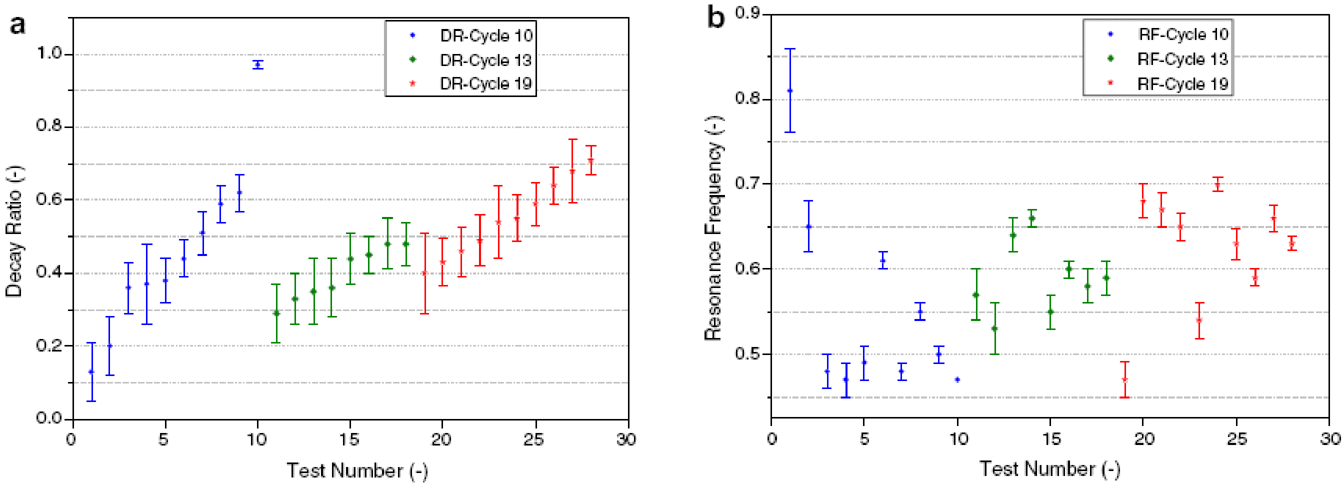

FIG. 9. Results for KKL stability tests in cycle 10, 13 and 19: (a) decay ratio and (b) resonance frequency, ref. [29].

Un-Planned events

<u>Forsmark-1 event</u>
Forsmark-1 has been in operation since 1980. Originally, the core design output was 2711 MWt. This was referred as "100% power". In 1986, the core design output was raised to 2928 MWt, an up-rating of 8%. Forsmark-1 is a member of the ABB Atom supplied BWR equipped with internal recirculation pumps. In each of the six ABB Atom "internal pump plants", the coolant flow is measured via instrumentation that monitors the inlet flows to eight core coolant channels. When Forsmark-1 was in the power ascension phase, after the annual shutdown of 1987 for refuelling and maintenance, oscillations of unusually large amplitudes were observed in the APRM signal, shortly after reaching 65% power [11].

LaSalle event

LaSalle Unit 2 underwent a dual recirculation pump trip event [12, 13]. After the pump trip, the unit experienced an excessive neutron flux oscillation under natural circulation conditions. In addition, as a result of the rapid power decrease, the feedwater heater level control system was unable to control the level in the feedwater heaters and began isolating the extraction steam from the heaters. This resulted in a positive reactivity addition because cooler feedwater was being supplied to the reactor. This, in turn, caused a power increase further reducing the margins from instability. Approximately seven minutes after the recirculation pump trip, the reactor automatically scrammed on APRM neutron high flux (118% trip). Although the power oscillations were larger than expected, no fuel thermal or mechanical limits were exceeded during the event.

WNP-2 event

WNP-2 is a GE BWR-6 plant. The concerned reactor core consisted of two fuel types having slightly different pressure drops. On August 15 1992, WNP-2 experienced power oscillations during the start-up operation [14]. About 33 hours before the event, the operators commenced a controlled power reduction from full power to 5%; afterwards the power was increased again up to 34% and then held constant for 3 hours. The operators continued the power increase along the 30% flow line up to about 36% power. The operators then began closing one of the two flow control valves in preparation for shifting the associated recirculation pump to fast speed. During this period, when power and flow decreased along the 76% rod line to power and flow levels of about 34% and 27%, respectively, oscillations occurred. Manual scram followed-up. The post event review showed in-phase oscillations, core wide, with peak to peak amplitude of about 25% of the rated power. The primary cause of the oscillation was considered to be the very skewed radial and bottom peaked axial power distribution (1.92 and 1.62 radial and average axial peaking factor, respectively).

5.2 Consequences on fuel and clad integrity

Thermal-hydraulic flow instability may cause large fluctuations in the global or local neutron flux in BWR cores. The resulting power oscillations, at a frequency 0.4 - 0.5 Hz in the fuel rods will, however, produce fuel temperature and heat rate changes which are considerably damped and delayed (due to the thermal inertia), typically by a factor >10 and 85-90° out-of-phase relative to the imposed heat generation rate.

An extensive data base was collected in relation to fuel rod behaviour following cyclic thermal loads at the Halden research reactor in Norway. As a main conclusion, it was found that there are large margins against cladding fatigue failure, partly due to the thermal damping by the fuel time constants. In addition, there is large margin to dryout as long as reverse flow is avoided following excessive amplitude oscillations; short dryout periods (cyclically followed by rewet) do not endanger the cladding integrity. The situation may be different under RIA (Reactivity Initiated Accident) conditions. Furthermore, recent experience shows that high burnup fuel may fail at energy releases as low as 30 Cal/g. This might have implications concerning the behaviour of high burnup fuel subjected to large amplitude oscillations.

The clad serves as a containment barrier to prevent radioactive fission products produced in the fuel from being released into the coolant. The cladding is subjected to rather severe stresses, both from the high-pressure coolant surrounding the fuel element, and by fission gas pressure and fuel swelling inside the fuel element. Above certain heat flux magnitudes, the heat transfer to the coolant will become unstable as a film of vapor forms to cover the fuel element surface. At this point the clad temperature will increase dramatically (several hundred degrees) leading to clad failure. This thermal limitation, known as the critical heat flux, is of primary concern in water-cooled reactor cores in which the coolant temperature is allowed to approach the boiling point.

6. STABILITY MONITORING AND LICENSING ASPECTS

As all areas in nuclear safety technology, there is a continuous feedback from operating experience, including stability events, to designers, to licensing authorities and vice versa (e.g. [22]). After investigation of the LaSalle 2 event by the licensee and further review by the NRC – Nuclear Regulatory Commission – (Bulletin N° 88-07) [23], US NRC the staff concluded that this event has significant generic safety implications.

6.1 Stability monitoring

Prevention, mitigation or suppression of the instabilities in a BWR plant constitute one of the safety operating requirement. So, instrumentation, mostly core related, plant control and protection systems, data interpretation and current strategies for prevention and mitigation, are considered. Among the implemented strategies, also as a follow up of specific requests by regulatory bodies, the "regional exclusion", the "quadrant APRM", the "power reduction", and the extended use of the core monitors, can be mentioned. The regional exclusion implies the introduction of a forbidden region in the operational power-flow map; in a few cases, an expanded 'alarm' region is also introduced. The quadrant APRM consists in allowing the possibility of generating control signals, including scram, from regional APRM related signals. The power reduction constitutes “an event oriented” strategy leading to automatic power reduction in the case of transients like pump trip and loss of feedwater pre-heaters.

Basically, a core (or stability) monitor consists or makes use of neutron detectors and signal preconditioning hardware, data sampling and recording, data analysis or process identification, estimation of the stability and data storing. An on-line feedback with the plant control system is avoided at the time being.

Examples are discussed in ref. [1], dealing with the use of stability monitors to detect sudden reductions of stability margins, to validate the exclusion region, to analyze stability test results, to follow the stability margins during an operational transient, to evaluate off-line stability, including the cases of core wide and regional oscillations, to predict stability boundaries (i.e. fixing the exclusion region) also as a function of fuel burn up.

Improvements in the performance of stability monitors can be envisaged in relation to the techniques for signal processing, i.e. noise analysis (limitations have been found in this area) and to the use as input signals of a larger number of plant signals including quantities like core flow and inlet subcooling, pump speed, feedwater flowrate and temperature.

As a general comment, it is noted that the number of LPRM is generally adequate for core stability monitoring. However, not all of these are directly and independently taken into account for deriving countermeasures. The recommendation is to give more emphasis to each LPRM, eventually improving the related reliability.

However, one of the several solutions designed is prevent operation in the region of the power-flow map most susceptible to oscillations using calculations to establish a region (the "exclusion region") on the power-to-flow map in which instabilities are credible.

6.2 Regulatory issues and requirements

The stability issue deserved a strict and constant attention by the regulatory bodies throughout all the years since the BWR design, e.g., the item "Suppression of Reactor Power Oscillations" constitutes the General Design Criterion 12 of the US NRC, 10 CFR 50. Renewed attention was given to the problem after the Caorso event (owing to regional oscillations), the La Salle event (owing to the large recorded APRM power) and the WNP-2 (owing to the event occurrence notwithstanding the countermeasures taken following La Salle).

From the industry side, the two main options adopted so far to comply with the licensing requests can be summarized by "prevention" and "detect and suppress" solutions. A so called long term solution, implying modifications to the reactor protection system is under consideration.

Regulatory requirements and recommendations include areas like training of operators, procedures, changes in the protection systems, stability predictions for new core loadings, stability measurements and core monitors installation and qualification. In addition, reliability of methods used in the past to prove stability is considered questionable; in a few cases the amplitude of power oscillations revealed higher than expected, out-of-phase oscillations might grow without causing scram, fuel failures might be a consequence of dryout following oscillations.

Definitely, the review of the stability issue by the licensing authorities, led to an independent evaluation of the design basis and of the protection measures to guarantee the fuel failures limits during instability events.

6.3 The exclusion region in the BWR operational flow-map

The "interim corrective actions" establish three regions on the power to flow map that restrict plant operation. In the Figure 10, regions A and B require immediate actions to exit the regions, excluding restarting a recirculation pump, ref. [1]. Plants without adequate protection against regional mode instability must immediately trip the reactor upon entry into Region A. Operation in Region C is only allowed during reactor start-up.

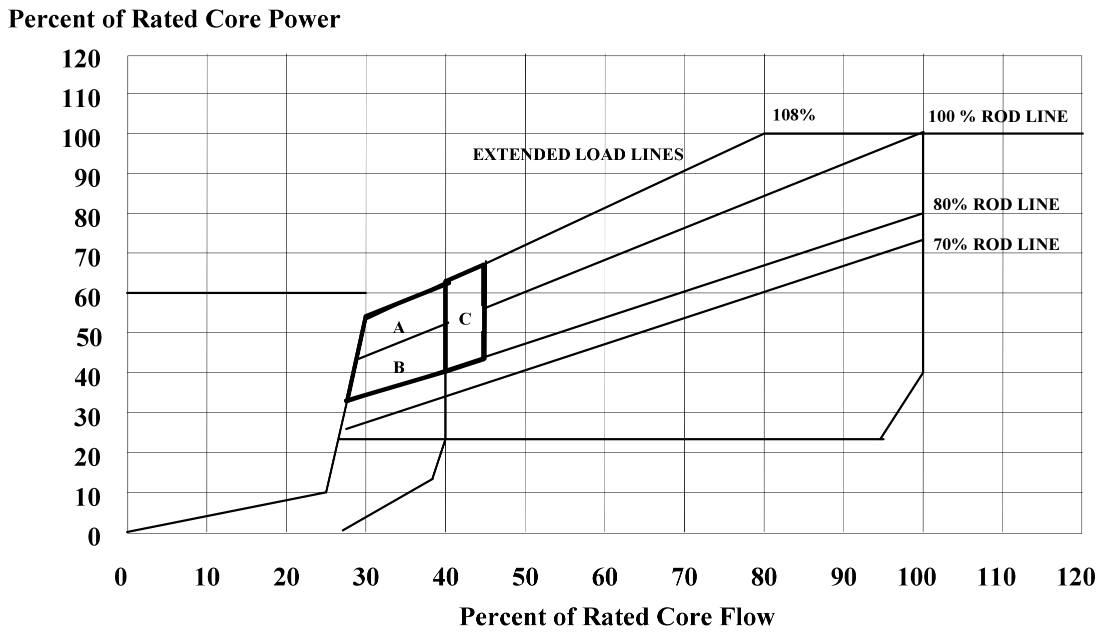

FIG.10. Defined operating regions. Region A: reactor power greater than 100% rod line, core flow less than 40% of rated core flow; region B: reactor power between 80% and 100% rod lines, core flow less than 40% of rated core flow; region C: reactor power greater than 80% rod line, core flow between 40% and 45% of rated core flow.

However, common practise in all countries is to avoid operating regions where power oscillations could occur. Because of the difference in the design of BWR there is a variety of requirements and recommendations that have been issued by licensing authorities. Current regulatory requirements and recommendations include the following items:

- training of operators to handle instability;
- procedures to handle instability;
- changes in protection system;
- prediction of stability for new core loading;
- measurements of stability at start-up after refuelling;
- stability monitors.

Examples of requirements and recommendations that have been issued in different countries are outlined in Tab. 3.

TABLE 3. REQUIREMENTS AND RECOMMENDATION TO AVOID REGIONS OF POWER OSCILLATIONS.

Country	**Requirements and recommendation**
Switzerland	• operator training in handling power oscillations; • restrictions in the operating region have been implemented; • stability monitors are used because increased surveillance is requested near the exclusion region.
Sweden	• have recommended that an overview of the operator training is to be made; • measurements on core stability must be performed when new fuel types are introduced.
USA	• procedural requirements must exist for initiating of manual scram under all operating conditions when all recirculation pumps trip or there are no pumps in operation.
Finland	• improvements in the reactor protection system to detect oscillations, both global and regional, have been required.
Germany	• automatic means of power limitation; • precalculations or measurements are carried out after refuelling. That also includes regions outside the operating region where the reactor could end up after a transient.
Japan	• all BWRs have implemented automatically selected rod insertion system and the operating region is defined to keep stable conditions; • stability is checked by means of code calculations before each new fuel cycle.
Finland	• the licensee is requested to define the limited stability values for power peaking factors and for control rod patterns; • core supervision system will give an alarm if these limits are exceeded.
Netherlands	• tests have been carried out to determine the stability of the Dodewaard reactor. The Dodewaard reactor is also equipped with an on-line monitor.
Spain	• current technical specifications of the Spanish BWR plants consider the stability domain on the core flow-power map subdivided into three regions. For example, the Region I (Exclusion), characterized by core flow below 40 % and thermal power over the 100 % load line, establishes that an immediate manual scram is required.

7. CONCLUSIONS

Instabilities that may occur during the BWR operation constitute a widely known problem in the scientific community addressed for more than thirty years. A great deal of literature is available including data and models. The analysis of the phenomena involved requires a multidisciplinary approach comprising various areas like transient thermal-hydraulics, neutron kinetics, fuel behavior including in-core fuel management, instrumentation, plant control and monitoring, and detailed knowledge of plant features. The use of large thermal-hydraulic system codes should be promoted in this area, provided 3-D neutron kinetics modeling and suitable numerics and specific user guidelines are implemented.

- All planned or unplanned instability events occurred in the region of low flow (around 30% core flowrate) and high power (around 50 %). In a number of situations scram occurred, only in one case was the high neutron flux scram set point (118% nominal power) reached based on the APRM signal: the LaSalle 2 event. In all cases the oscillation frequency was around 0.5 Hz. Preventive and mitigative approaches have been used so far to deal with potential instability situations. Either systematic plant tests aiming at the identification of stability boundaries, or specifically developed monitoring techniques, or both of these, constituted the way to prevent undesired challenges to the scram system. In the area of monitoring, wide progresses have been observed as a consequence of the availability of powerful computers and of sophisticated techniques for noise analysis. An adequate number of LPRM transducers is available in each BWR plant.

Finally, the main conclusions and recommendations can be summarized below:

- axial and radial power distributions have a strong influence on stability: the variation of these during the reactor life or during a transient may bring the core in a configuration prone to instability;
- scram and coolant flowrate increase (when FW is available at nominal conditions), slight changes of control rod positions (including SRI) are typical countermeasures that can be taken, should an instability event originate;
- reference experiments in qualified facilities and/or plant data should be used to develop a common understanding about suitability of predictive tools covering areas like user effect, uncertainties in boundary conditions, optimization of nodalization, capabilities of numerics;
- the sophistication and the qualification levels of core monitors could be largely improved and demonstrated, respectively;
- the execution of experiments in loops of different complexity allowed the identification and characterization of different types of instabilities;
- experiments from out of pile facilities are necessary to qualify thermalhydraulic codes;
- the coupled codes method, which consists in incorporating three-dimensional (3D) neutron modeling of the reactor core into system codes, is particularly suited for simulating transients that involve core spatial asymmetric phenomena and strong feedback effects between core neutronics and reactor loop thermal-hydraulics;
- the use of calculations to establish a region (the "exclusion region") on the power-to-flow map in which instabilities are credible is essential to prevent operation in the region of the power-flow map most susceptible to oscillations;
- the review of the stability issue by the licensing authorities, led to an independent evaluation of the design basis and of the protection measures to guarantee the fuel failures limits during instability events;

- current regulatory requirements and recommendations to avoid power oscillations regions include: training of operators to handle instability, procedures to handle instability, changes in protection system, prediction of stability for new core loading and measurements of stability at start-up after refuelling.

REFERENCES

[1] F. D'Auria, W. Ambrosini, T. Anegawa, J. Blomstrand, J. In De Betou, S. Langenbuch, T. Lefvert, K. Valtonen, 1997, State of the Art Report on Boiling Water Reactor Stability (SOAR on BWR), *OECD-CSNI report OECD/GD (97) 13*, Paris.

[2] P. Hänggi, 2001, Investigating BWR Stability with a New Frequency-Domain Method and Detailed 3D Neutronics, *Doctor Dissertation*, S. F. Institute of Technology Zurich, Switzerland.

[3] EURATOM, 1967, Proceedings of "Symposium on Two-Phase Flow Dynamics" EURATOM, EUR 4288e, Vols. I and II, 4 - 9 September, Eindhoven, Holland.

[4] F. D'Auria, V. Pellicoro, 1996, Local Instability in BWR Reactor Simulator**,** *ICONE-4 Conference*, New Orleans, USA.

[5] J. L. Muñoz-Cobo, S. Chiva, A. Sekhri, 2004, A Reduced Order Model of BWR Dynamics with Subcooled Boiling and Modal Kinetics: Application to Out of Phase Oscillations, *Annals of Nuclear Energy*, **31**, pp. 1135-1162.

[6] A. Bousbia-Salah, F. D'Auria, M. Bambara, 2004, Sensitivity Analysis of the Peach Bottom Turbine Trip 2 Experiment, *Kerntechnik*, **69**, pp. 7-14.

[7] L. A. Carmichael and R. O. Niemi, 1978, Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2, *EPRI Report NP-564*.

[8] Y. Waaranpera, S. Anderson, 1981, BWR Stability Testing: Reaching the Limit Cycle Threshold at Natural Circulation**,** *Trans. Am. Nucl. Soc*., **39**, pp. 868.

[9] A. A. Sandoz, S. F. Chen, 1983, Vermont Yankee Stability Tests during Cycle 8**,** *Trans. Am. Nucl. Soc.*, **45**, pp. 727.

[10] D. Dayal, G. Preusche, L. P. Brandes, M. Frank, H. Heckermann, R. Seepolt, K. P. Wachter, 1987, Stability Tests at KWU Nuclear Power Plants**,** *Proceedings of ANS Topical Meeting on Anticipated and Abnormal Transients in Nuclear Power Plants*, April 12-15, Atlanta, USA.

[11] J. Blomstrand, A. Spencer, I. Ganea, L. Ekstam, P. Reibe, K. Adielsson, F. Schwartz, C. Kvarnström, L. E. Häll, 1988, Noise Analysis of Core Coolant Channel Flow Signals, recorded in Swedish and Finnish BWRs, *Progress in Nuclear Energy*, **21**, pp. 777-778.

[12] US NRC, 1988, Augmented Inspection Team Report (AIT) Nos. 50-373/88008 and 50-374/880088.

[13] W. Wulff, H. S. Cheng, A. N. Mallen, U. S. Rohatgi, 1992, BWR Stability Analysis with the BNL Engineering Plant Analyzer**,** *NUREG/CR-5816, BNL-NUREG-52312*.

[14] US NRC, 1992, Augmented Inspection Team (AIT) Report of WNP-2 No. 50-397/92-30.

[15] A. L. Costa, F. Maggini, A. Bousbia-Salah, F. D'Auria, 2004, Peach Bottom Cycle 2 Stability Analysis Using RELAP5/PARCS, *UIT 2004*, Genova.

[16] J. Solís, M. N. Avramova, K. N. Ivanov, 2002, Temporal Adaptive Algorithm for TRAC-BF1/NEM/COBRA-TF Coupled Calculations in BWR Safety Analysis, *Annals of Nuclear Energy*, **29**, pp. 2127-2141.

[17] W. L. Weaver, E. T. Tomlinson, D. L. Aumiller, 2002, A generic semi-implicit coupling methodology for use in RELAP5-3D, *Nuclear Engineering and Design*, **211**, pp. 13-26.

[18] R. Miró, D. Ginestar, G. Verdú, D. Hennig, 2002, A Nodal Modal Method for the Neutron Diffusion Equation. Application to BWR Instabilities Analysis, *Annals of Nuclear Energy*, **29**, pp. 1171-1194.

[19] A. Hotta, M. Honma, H. Ninokata, Y. Matsui, 2001, BWR Regional Instability Analysis by TRAC/BF1-ENTRÉE-I: Application to Density-Wave Oscillation, *Nuclear Technology*, **135**, pp. 1-16.

[20] A. Hotta, H. Ninokata, 2002, Numerical Study on Observed Decay Ratio of Coupled Neutronic-Thermal Hydraulic Instability in Ringhals Unit 1 under Random Noise Excitation, *Journal of Nuclear Science and Technology*, **39**, pp. 138-149.

[21] A. Hotta, T. Anegawa, T. Hara, H. Ninokata, 2003, Simulation of Boiling Water Reactor One-Pump Trip Transient by TRAC/BF1-ENTRÉE, *Nuclear Technology*, **142**, pp. 205-229.

[22] E. Laurence, L. Phillips,1990, Resolution of US Regulatory Issues Involving Boiling Water Reactor Stability, *Proceedings of the OECD/CSNI International Workshop on Boiling Water Reactor Stability*, October 17-19, Brookhaven, Holtsville NY, CSNI Report 178.

[23] US NRC, 1988, Power Oscillation in Boiling Water Reactors (BWR)**,** *NRC Bulletin No. 88-07*, Washington, USA.

[24] F. D'Auria, A. Del Nevo, N. Muellner, 2008, Insights Into Natural Circulation Stability, Updated, *IAEA Course on Natural Circulation in Water-Cooled Nuclear Power Plants, International Centre for Theoretical Physics (ICTP), Trieste, Italy 23-27 June Paper ID T07*.

[25] F. D'Auria, A. Bousbia-Salah, 2008, Coupled 3d Neutron Kinetics And Thermal-Hydraulics Techniques And Relevance For The Design Of Natural Circulation Systems, Updated, *IAEA Course on Natural Circulation in Water-Cooled Nuclear Power Plants, International Centre for Theoretical Physics (ICTP), Trieste, Italy 23-27 June Paper ID T11*.

[26] A. Bousbia-Salah, F. D'Auria, 2006, Analysis of the Peach Bottom flow stability test number 3 using the coupled RELAP5/PARCS code, *J. Annals of Nuclear Energy, Vol.* ***33****, pages 646–652.*

[27] H-T. Lin, J-R. Wang, C-L. Hsieh, C. Shih, S-C. Chiang, T-L. Weng, 2006, Kuosheng BWR/6 stability analysis with LAPUR5 code *J. Annals of Nuclear Energy, Vol.* ***33****, pages 289-299.*

[28] A. Dokhane, D. Hennig, Rizwan-uddin, R. Chawla, 2007, Interpretation of in-phase and out-of-phase BWR oscillations using an extended reduced order model and semi-analytical bifurcation analysis, *J. Annals of Nuclear Energy, Vol.* ***34****, pages 271-287.*

[29] A. Dokhane, H. Ferroukhi, M.A. Zimmermann, C. Aguirre, 2006, Spatial and model-order based reactor signal analysis methodology for BWR core stability evaluation, *J. Annals of Nuclear Energy, Vol.* ***33****, pages 1329-1338.*

APPENDIX A1

SELECTED OUT-OF-PILE TEST FACILITIES

A.1.1 The FRIGG loop

The FRIGG loop was built by ASEA in the mid-1960´s, and has been in operation since 1966. It is located in the ABB Atom Laboratories in Västerås, Sweden. Although its original design power was 8 MW, the loop was actually operated up to 9 MW occasionally during the late 1960´s.

The test channels were equipped with an abundance of instrumentation, providing a wealth of detailed information. Three tests series were made for the 36-rod cluster geometry. Two of them are of interest, as regards flow stability:

i) FRIGG - 3 (channel FT 36 b). This test assembly combined a uniform axial power distribution with a non-uniform radial power distribution (which was varied to simulate the radial shape of the neutron flux depression in the fuel assemblies of the Marviken core). These measurements are of special interest in the context of flow stability, since a void impedance gauge had been installed at the exit of the test section which enhanced the two-phase pressure drop (detrimental to stability). Periodic flow oscillations were observed in several of the operating situations studied.

ii) FRIGG - 4 (channel FT 36 c). This test assembly made use of the same radial power distribution as in FRIGG - 3, whereas the axial power distribution was of the ”chopped cosine” type. This test series also made use of the above mentioned instrumentation at the channel exit. Although no periodic oscillations were encountered, the instability threshold was established via extrapolation of the flow noise.

The experience from the FRIGG - 3 and FRIGG - 4 tests has been used extensively for benchmarking of a variety of dynamic codes, for example various versions of RAMONA.

A.1.2 The Karlstein loop

This 15 MW loop has been designed and built by Siemens KWU. The facility, which has been operating since 1988, is located at the Siemens laboratories at Karlstein, in Germany. It is dedicated to the thermal-hydraulic development and testing of electrically heated mockups of fuel assemblies for both BWRs and PWRs. For BWR applications, the loop is fully qualified for testing of full-scale fuel assemblies.

The tests were conducted at natural circulation, in order to realise the “parallel channel boundary condition”. The data recorded comprise: operating pressure, inlet subcooling, coolant mass flow at inlet and outlet of test section, surface temperatures of heater rods, differential pressures (downcomer, inlet, exit), etc.

The measurements were initiated at stable conditions, with fixed power and subcooling. For assessing the detailed stability characteristics, pseudo-random binary signals (PRBS) perturbations of the power input to the heated rods were imposed - for half an hour. Their amplitude was chosen so as to provide a significant signal response, while remaining within the linear domain. The decay ratio was then evaluated from the signal recordings.

This procedure was repeated as the bundle power was increased stepwise, in a gradual approach to the stability threshold. Further power increase was found to produce limit cycle oscillations. Their amplitudes turned out to depend on the power level. These power increases were allowed to continue, until periodic dryout and rewetting was observed. This procedure made it possible to establish the threshold power for boiling transition during flow oscillations. No data is available from this facility.

A.1.3 The ATLAS loop

This 17 MW full-scale thermal-hydraulic test facility for BWR fuel assemblies has been built by GE. It is located in San Jose, California.

ATLAS was built as a separate effects not as a system test facility. Thus, it was not configured for performing stability tests for single fuel bundles. Due to the complexity inherent in the BWR stability process (Parallel channel interactions, neutronic gain, excitation of higher harmonic modes, sensitivity to reactor operating state parameters such as core power distribution effects, etc.) GE's approach has been to perform stability testing of actual operating BWR's and to qualify computation models to the reactor test data which includes the coupled thermal hydraulics and neutronic phenomena. GE believes that the best approach for understanding and designing for BWR fuel stability is by reactor qualification of very sophisticated computer codes. Of course all the fuel component pressure drop loss coefficient data generated at ATLAS were a vital part of the stability assessment process.

A.1.4 The NFI loop

This 1.8 MW NFI loop was designed and built by NFI. The loop has been operating since 1993 and it is located at NFI Tokai-mura. The loop is designed for high temperature steam and water two-phase flow tests. The maximum working pressure and temperature are 9.8 MPa and 580 K, respectively. The large volume of the steam separator/condenser provides stable loop operation during the flow oscillating stability test. The total capacity of 1.8 MW dc electric power from the two rectifier units is shared by two test channels. The thermal-hydraulic testing that has been conducted so far includes single and two-phase pressure drop tests, flow-induced rod vibration tests, dry-out and post dry-out tests, and thermal-hydraulic stability tests in the parallel channel configuration.

The parallel channel test parameters included variations of flow rate, inlet, subcooling, inlet orifice, size, and axial power profile, i.e. uniform, chopped cosine and bottom peaked power distributions. The stability threshold was measured by increasing the test channel power until the amplitude of the inlet flow oscillation became larger than the noise level. Beyond the threshold, the inlet flow exhibited typical density wave oscillation with frequency of 0.2 to 0.6 Hz. This stationary limit cycle oscillation extended the flow amplitude to several tens of a percent at which periodic dry-out and rewetting were observed by the heater temperature measurements.

The test data were analyzed by the NFI stability analysis code STAIF-PK. The predicted threshold powers were always below the measured data. Agreement between data and predictions of the natural frequency was good. It was concluded that the conservative predictions by the STAIF-PK code imply a certain margin of the channel stability for the newly designed 9x9 fuel, since the stability performance of this fuel has been designed using the STAIF-PK code.

Among various results, it was found that a subcooling increase causes a decrease in the power threshold for oscillations, this effect is less pronounced when a high pressure drop inlet orifice is adopted.

A.1.5 The BEST loop

The acronym for BEST is BWR Experimental loop for Stability and Transients tests. This test facility was built by TOSHIBA in the early 1980´s. The mechanism operating conditions for the loop are 10.3 MPa (fluid pressure), 538 K (fluid temperature), 6 MW (heated bundles power) and 17 kg/s (bundles flowrate). The facility can be used for investigating both steady-state and transient characteristics of various thermal-hydraulic conditions that may be encountered in BWR operation, for example: unusual combinations of high power and reduced coolant flow, thermal-hydraulic stability, exceptional transients, boiling transition, post-dryout, etc. Testing can be made of full-length bundles of BWR fuel assemblies, of reduced width.

In the early 1980´s, stability tests were conducted which involved 2x2 lattice bundles: representing a reduced width version of an 8x8 lattice assembly. The stability tests which have been conducted in recent years have involved a 3x3 lattice bundles, a reduced width version of an advanced High-Burnup 9x9 lattice assembly design (STEP-III type A). A substantial flow bypass was used in all these tests, to realise the "parallel channel condition".

As regards the above mentioned advanced High Burnup fuel STEP III, there are two design versions: type A and type B. The thermal-hydraulic testing of the type A version of this fuel design was made in the BEST loop. The stability tests made use of a bypass section which was heated by an 8x8 lattice bundle. The actual tests addressed conditions up to the instability threshold, comprising investigations of the effects on stability of parametric changes in inlet mass flow, inlet subcooling and axial power profile, as well as the inclusion in the test bundle of part-length fuel rods.

A.1.6 The PIPER-ONE loop

Under the sponsorship of ENEA, the DCMN of the Pisa University (in Italy) commenced the design and construction of this loop in 1981. It is located at the Scalbatraio Laboratory, in the vicinity of Pisa. It was first operated in 1985.

The loop is an integral facility designed for reproducing the behaviour of BWR thermal-hydraulics in operating situations which are dominated by gravity forces. The main objective of the research is to provide experimental material that can be used to qualify computer codes for thermal-hydraulic analysis for simulating a variety of operating situations, for example flow stability.

The coolant channel was designed to permit testing of a full-length 4x4 lattice rod bundle also including the unheated length. The maximum power of the loop is slightly more than 300 kW, corresponding to about 30% of the nominal linear power of the reference BWR. The rods are indirectly heated with chopped cosine axial power distribution. The loop operates at natural circulation (which is convenient for flow stability testing) and can be used up to pressures of 10 MPa.

Instabilities have been analyzed by transforming into the frequency domain the time trends signals got from the experiments, the FFT (Fast Fourier Transform) was applied to the signals from pressure transducers (7 of these are installed throughout the core simulator).

A "quasi" stable limit cycle was observed by cross correlating DP signals along the core. The fundamental frequency of the oscillations was found at about 0.5 Hz. Unsteady dryout and rewet cycles were also observed.

A more recent investigation, led to characterize the "punctual" instability. Essentially, the use of the FFT technique showed that in some situation, instability does not propagate along the channel axis. Amplitude of oscillations is close to zero at the channel inlet and outlet but it peaks around the zone of peak power. Thermal-hydraulic mechanisms for rise and smoothing of instabilities appear to exist. 3D phenomena inside the bundle and friction could be important in this respect.

A.1.7 The FIST loop

In the early 1980's, a series of natural circulation tests was conducted in the FIST facility. They were done covering a wide range of powers, water levels and system pressures. This facility is a full height and full pressure BWR test loop. It was built in the late 1970's, and is located in San Jose, California. It was designed to investigate system responses to various transient events, including stability. In fact, FIST is an acronym for Full Integral Simulation Test. The facility is volume scaled by a ratio of 1/624 to a BWR/6 plant containing 624 fuel assemblies (such as Cofrentes, in Spain). The core is represented by an 8x8 electrically heated bundle.

In more detail, the facility embodied an accurate simulation of the complete BWR recirculation system, including two small functional jet pumps, full height test vessel and internals, and correctly scaled fluid volume distribution and flow resistances. The heated fuel bundle (including rod power distributions, spacers, zircaloy fuel channel, upper and lower tie plates) was faithfully simulated, and the same applied to the bypass region and the fuel leakage paths.

APPENDIX A2. IN-CORE EXPERIMENTS

A.2.1 The Halden Reactor

The OECD Halden Project operates a small (20 MW) natural circulation BHWR, cooled and moderated by heavy water. It is located at Halden, in southern Norway, and was built in the late 1950´s. The core is of the open-lattice type, containing about 100 fuel assemblies - all located within individual coolant channels which are rather widely spaced out. As regards stability, the following comments are pertinent:

i) The coolant channels are free-standing without any supply tubes (where single-phase inertial effects might influence the stability characteristics).

ii) Since the heavy water is close to saturation, there are virtually no effects from inlet subcooling on stability.

iii) Almost all the moderation takes place in the heavy water located in between the channels. It follows that void variations which may occur within the coolant channels do not produce any significant reactivity feed-back via the fuel on the channel flow stability. Under these conditions, any periodic channel flow fluctuations which may appear are of purely thermal-hydraulic nature.

The issue of flow stability in the Halden Reactor was first investigated in 1964. These initial studies led to the design of an advanced test assembly to be used for more detailed investigations of the power removal capacity of a coolant channel in that reactor, as regards both dryout and stability. These test assemblies differed from the normal assemblies used in the reactor, as regards design, fuel weight and enrichment. The test channel was equipped with:

i) an externally operated orifice at the channel inlet, making it possible to study the influence of the inlet throttling on flow stability over a very wide orificing range;

ii) extra instrumentation at the channel exit. The corresponding pressure loss came to exercise a significant influence on the stability characteristics of the channel flows.

A.2.2 The Nordic BWR of design generation with internal pumps

Six of the Nordic BWRs are equipped with instrumentation for monitoring inlet flows to a number of core coolant channels. In these reactors, it is possible to make simultaneous observations of:

i) channel flow fluctuations of purely thermal hydraulic nature (via the flow monitoring instrumentation);

ii) coupled flow fluctuations which involve neutron feedback via the fuel.

Some experience from studies of these phenomena were conducted around the mid-1980´s. They refer to conventional 8x8 lattice fuel, as well as the modified design of the same lattice fuel called SVEA-64 which was being introduced in some of these reactors at that time.

As regards the 8x8 lattice fuel, analyses of signal recordings that had been made in several of these plants showed that the two oscillatory phenomena mentioned above coexisted in the core, but appeared at different frequencies and had different damping characteristics. For example, at the minimum core coolant flow permitted at full power, the channel flows were found to fluctuate incoherently - and significantly faster (at about 1 - 1.1 Hz) than the neutron flux (at about 0.75 Hz), which indicates the frequency of the coherent components in the channel flow fluctuations. It was observed that the flow fluctuations through the individual channels were strongly coupled to the corresponding channel power, the flow noise levels depending almost linearly on the channel power. During operation at stretch-out, the frequency of the flow fluctuations gradually increased up to about 1.4 Hz, which was reached at the upper end of the permitted core coolant flow range, at full power.

Conversely, when the core power was lowered via the core coolant flow along the flow control line (from full power and minimum permitted coolant flow), the frequency of the flow fluctuations was reduced. At 75 % power, for example, their frequency had fallen to 0.7 Hz - while the flow noise levels had increased.

In the operating situation studied, it was found that while the neutron flux had oscillated at 0.49 Hz with a decay ratio of 0.8, the four individual channel flows involved in the recordings had fluctuated at characteristic frequencies that ranged from 0.43 Hz up to 0.56 Hz, with decay ratios lower than 0.3. In this operating situation, the neutron flux and channel flows had reached frequencies that were at least of a similar order of magnitude, even though their damping properties differed significantly. Coherence studies showed that the channel flows had fluctuated independently of each other, and even of the neutron flux - despite the rather high decay ratio for the core.

APPENDIX B. CODES MORE FREQUENTLY REPORTED IN THE LITERATURE CONCERNING BWR STABILITY

B.1 Frequency-Domain Codes

TABLE B.1. CHARACTERISTICS OF FABLE/BYPASS CODE.

Code	**FABLE/BYPSS**
Property/ Developer	• General Electric, USA
Capabilities	• 1D parallel channel (plus bypass) thermal-hydraulic simulation of the core • Point neutron kinetics • Calculation of the Decay Ratio • Simplified ex-core modeling • Adopted to confirm that new fuel design meet NRC acceptance criteria and define boundaries of exclusion region
Thermal-hydraulics	Multi-channel + bypass model; Mixture equations with subcooled boiling; Bankoff slip correlation
Neutronics	Point neutron kinetics with void reactivity feedback; Void feedback based on flux squared weighting of nodal void reactivities from PANACEA 3D wrapup data; Axial variation in parameters such as control fraction not accounted for; Method of evaluating void coefficient tied to qualification basis and application procedure

TABLE B.2. CHARACTERISTICS OF HIBLE CODE.

Code	**HIBLE**
Property/ Developer	• Hitachi, Japan
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Point neutron kinetics • Calculation of the Decay Ratio by linear Laplace transformed equations • Simulation of different BWR types: jet pumps and internal pumps • Simulation of pressure tube type reactors • Simulation of experimental test loops
Thermal-hydraulics	Three equation slip flow models; Options for two-phase flow correlations; Up to 20 channel groups
Neutronics	Point neutron kinetics with six delayed neutron groups; Coolant density and Doppler reactivity considered.

TABLE B.3. CHARACTERISTICS OF K2 CODE.

Code	**K2**
Property/ Developer	• Toshiba, Japan
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Point neutron kinetics • Calculation of the Decay Ratio • Simulation of different BWR types: jet pumps and internal pumps • Simulation of experimental test loops • Licensing code for TOSHIBA BWR
Thermal-hydraulics	Three equation drift-flux model for parallel channels; Options for physical law from design correlations: void-quality, subcooled boiling, Δp; Several channel groups; one group for bypass; Linearization of the equations and Laplace transformation
Neutronics	Point neutron kinetics with six delayed neutron groups; Laplace transformation of basic equations; Neutron flux transfer functions for fundamental and higher mode; Higher mode transfer function calculated from the higher mode subcriticality obtained by a 3D code; Transfer functions of fundamental an higher modes used for core-wide and regional oscillations respectively.

TABLE B.4. CHARACTERISTICS OF LAPUR-5 CODE.

CODE	**LAPUR-5**
Property/ Developer	• Nuclear Regulatory Commission, USA • ORNL, USA
Capabilities	• Core-wide, out-of-phase, and channel decay ratios from calculated frequency-domain transfer functions • 1D parallel channel (maximum of 7) TH simulation of the core with dynamic flow redistribution • Full Laplace domain (i.e. not just along the imaginary axis) pole search for increased decay ratio estimate accuracy
Thermal-hydraulics	Two-fluid slip model in the core; Non-equilibrium (subcooled boiling) model; Integral formulation of ex-core vessel components (recirculation loop); Core pressure/flow boundary conditions are automatically selected to estimate the core-wide or out-of-phase oscillation modes.
Neutronics	Point kinetics for core-wide neutronics; Out-of-phase mode neutronics modelled using the point kinetics equivalent equations for the first subcritical mode; Reactivity feedback calculated as power-square average of "local" reactivity contributions; Density reactivity coefficient may be input as a polynomial of local density or it may be calculated by LAPUR-5 based on 2-group cross sections and control-rod positions.

TABLE B.5. CHARACTERISTICS OF NUFREQ CODE.

Code	**NUFREQ**
Property/ Developer	• R.T. Lahey, G. Yadigaroglu (General Electric)
Capabilities	• Simulation of simple two-phase sections or loop • Point neutron kinetics
Thermal-hydraulics	Homogeneous equilibrium flow for a single channel; Δp adjusted with correction factors; Linearization of the equations and Laplace transformation
Neutronics	Simple point neutron kinetics; Reactivity coefficients for void and Doppler; Linearization of the equations and Laplace transformation.

TABLE B.6. CHARACTERISTICS OF NUFREQ-NP CODE.

Code	**NUFREQ-NP**
Property/ Developer	• RPI, USA
Capabilities	• 1D parallel channel TH simulation • Coupling with point, 1D, 2D or 3D neutron kinetics models • Ex-core recirculation dynamics
Thermal-hydraulics	Three balance equations in the liquid region of the channel; Four equations drift-flux model with thermal non-equilibrium in the subcooled boiling region; Thermal equilibrium drift-flux model in the bulk boiling region.
Neutronics	Coupling with models for point, 1D, 2D or 3D kinetics

TABLE B.7. CHARACTERISTICS OF ODYSY CODE.

Code	**ODYSY**
Property/ Developer	• General Electric, USA
Capabilities	• 1D parallel channel thermal-hydraulic core simulation • 1D neutron kinetics or point kinetics with higher harmonics • Calculation of Decay Ratio • Qualified for GE internal evaluation of new fuel design
Thermal-hydraulics	Multiple channel types; Five equation drift flux model (two mass and energy equations, plus a single momentum equations); In-channels and Bypass Regions; Transient redistribution of flows due to channel coupling; Direct moderator heating; Best estimate thermal-hydraulics correlations.
Neutronics	1-D kinetics with void and Doppler reactivity feedback; Kinetics parameters collapsed form 3D PANACEA wrapup at specific plant conditions; Flux squared weighting of kinetic parameters; Impact of axial and radial variations in control fraction and power shape accounted for; Sets of specific collapsed parameters for each fuel type, control and power distributions in axial nodes; Fitting of coefficients describing the effect of fuel temperature and moderator density considering the various channel groups.

TABLE B.8. CHARACTERISTICS OF STAIF CODE.

Code	**STAIF**
Property/ Developer	• Siemens-KWU, Germany
Capabilities	• 1D parallel channel thermal-hydraulic core simulation • 1D neutron kinetics • Calculation of Decay Ratio for fundamental mode and the higher harmonics • Siemens design tool for linear stability
Thermal-hydraulics	Multiple channel types with independent geometry and axial power distributions; 2 mass (vapour, mixture), 2 energy (vapour, mixture) and one momentum (mixture) equations; Subcooled boiling consideration; In-channel and bypass flow simulation; Fraction of heat generated directly in the moderator including unheated parts; Up-to-date design basis thermal-hydraulic and pressure drop correlations.
Neutronics	1 group diffusion theory with 1D axial variation of neutron flux; 1D axial void and Doppler reactivity feedback; Influence of the control rod pattern by control rod dependent properties; 6 delayed neutron groups for Uranium and MOX Fuel; Data transfer from 3D core-simulator for accounting specific plant conditions; Determination of higher harmonics.

TABLE B.9. CHARACTERISTICS OF STAIF-PK CODE.

Code	**STAIF-PK**
Property/ Developer	• Nuclear Fuel Industries Ltd., Japan
Capabilities	• One-dimensional parallel channel simulation of the BWR core • Point kinetics • Coupled thermal-hydraulic and neutronic core stability • Reactivity feedback associated with fundamental or higher harmonic modes • Calculation of decay ratio
Thermal-hydraulics	Multiple channel types with distinct geometry and power distributions; One dimensional two-phase drift-flux model; Five equations: 2 mass and 2 energy (vapour, mixture), 1 momentum; In-channel and bypass flow; Subcooled boiling model and direct moderator heating models.
Neutronics	Point kinetics for critical and subcritical mode; Six group delayed neutron equation; Void reactivity as a function of void fraction and control rod density for each node; Reactivity feedback with squared nodal power (or flux) weighting for the fundamental or higher harmonic mode.

B.2 Time Domain Codes

TABLE B.10. CHARACTERISTICS OF ATHLET CODE.

Code	**ATHLET**
Property/ Developer	• Gesellschaft fur Anlagen- und Reaktorsicherheit (GRS) mbH, (Germany)
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Neutronic described by point or 1D kinetics • Detailed description of the whole plant, including BoP
Thermal-hydraulics	1D flow models based on drift-flux and thermal non-equilibrium; six equation two-fluid model under development; Fully implicit time-integration with accuracy controlled time-step size
Neutronics	Point kinetics or 1D kinetics with six delayed neutron groups; Coupling with 3D neutronics model QUABOX/CUBBOX under development

TABLE B.11. CHARACTERISTICS OF DYNAS-2 CODE.

Code	**DYNAS-2**
Property/ Developer	• Nuclear Fuel Industries Ltd., Japan
Capabilities	• One-dimensional parallel channel simulation of the BWR core • Three-dimensional kinetics • Coupled thermal-hydraulic and neutronic core stability • Calculation of LPRM and APR signals • Calculation of Critical Power Ratio
Thermal-hydraulics	Multiple channel types with distinct geometry; One-dimensional two-phase drift-flux model; Five equations: 2 mass and energy (vapour and mixture), 1 momentum (mixture); In-channel and bypass flow; Subcooled boiling and direct moderator heating models.
Neutronics	One-group diffusion model; One node for each bundle and 24 axial nodes; Six-group delayed neutron equation; Void reactivity and Doppler reactivity feedback as a function of void fraction and control rod density for each node

TABLE B.12. CHARACTERISTICS OF DYNOBOSS CODE.

Code	**DYNOBOSS**
Property/Developer	• Instituto de Estudos Avançados (Brazil), RPI (USA)
Capabilities	• 1D parallel channel thermal-hydraulics • Point neutron kinetics • Vessel component modeling
Thermal-hydraulics	Four equation drift-flux model; Vapour phase always in saturated conditions; subcooled boiling models: mechanistic and profile fit; Zuber-Findlay drift-flux model; fluid thermophysical properties evaluated on the basis of an average system pressure; Liquid density taken at saturation; Semi-implicit numerical method, with capability to choose time discretization (explicit, Crank-Nicholson, implicit); Iterative procedure to attain steady-state conditions.
Neutronics	Point neutron kinetics; 2D one speed space dependent steady-state calculations; Either cylindrical or rectangular geometry; Reflection factors at the core top, bottom and lateral surfaces; Linear feedback for Doppler and void effect; Successive Line Overrelaxation (SLOR) method for matrix inversion.

TABLE B.13. CHARACTERISTICS OF EPA.

Code	**BWR-EPA** (with HIPA-BWR Code)
Property/ Developer	• NRC and BNL, USA
Capabilities	• 1D nonhomogeneous, nonequilibrium two-phase flow in vessel • Point kinetics with approximation for 1D, time-dependent power shape • Fixed nodal representation of vessel (55 nodes), 3 core channels • Dynamic ex-vessel models for BoP, suppression pool, wetwell, drywell and control systems • Interactive on-line graphics and acceptance of control and failure signals from keyboard via telephone line • Four times faster than real-time simulation speed
Thermal-hydraulics	Four equations drift flux model with nonequilibrium phase change; Loop momentum balance and mixture volumetric flux divergence equations; Core-average, hot and bypass channels with 12 nodes each; Nonequilibrium boiling, flashing and condensing in the vessel, equilibrium phase change in containment for relative humidity; wall shear and form loss according to HTFS; Polytropic steam flow through turbines (power and FW pump drive, with speed-dependent isentropic turbine efficiency).
Neutronics	Point kinetics, six delayed neutron groups;Reactivity feedback for Doppler, moderator temp

TABLE B.14. CHARACTERISTICS OF PANTHER CODE.

Code	**PANTHER**
Property/ Developer	• Nuclear Electric, United Kingdom
Capabilities	• 1D thermal-hydraulics • 3D neutronic representation of the core • Applicable also to PWR, VVER, RBMK, AGR and MAGNOX
Thermal-hydraulics	1D drift flux model with EPRI slip correlation; Subcooled boiling model; Tabular dependence of coolant thermophysical properties; Also coupled to VIPRE-01
Neutronics	Rectangular or hexagonal assemblies; Sector symmetries and range of boundary conditions; Simple mesh refinement consistent 3D to 1D condensation; Steady-state and transient analytic nodal solution of diffusion equations; Multigroup (up to 12 groups); Time differencing based on stiffness confinement; Nodal pin-power reconstruction; Extensive range of steady-state search options and transient drivers; Representation of within-node burn-up variation and rod tip treatment; Steady-state and transient thermal and poison feedback; Transient decay heat representation; Extensive tabular dependence of cross-sections on temporal and history parameters.

TABLE B.15. CHARACTERISTICS OF QUABBOX/CUBBOX-HYCA CODE.

Code	**QUABOX/CUBBOX-HYCA**
Property/Developer	• Gesellschaft fur Anlagen- und Reaktorsicherheit (GRS) mbH, (Germany)
Capabilities	• 3D neutronic representation of the core • 1D thermal-hydraulics
Thermal-hydraulics	1D-single and two-phase flow model using drift-flux correlations; Channel by channel representation; Core boundary conditions either specified at core inlet or pressure drop along the core
Neutronics	Full 3D two-group coarse mesh solution of neutron diffusion equations based on local polynomial flux expansion for steady-state and transient conditions; Time integration including frequency transformation; Flexible interface to cross-section library describing dependence on all relevant parameters.

TABLE B.16. CHARACTERISTICS OF RAMONA-3 CODE.

Code	**RAMONA-3**
Property/ Developer	• ScandPower, Norway
Capabilities	• 1D thermal-hydraulic simulation of vessel components with parallel channels in the core • 3D neutronic representation of the core with 6x6 nodes • Ex-core component modeling
Thermal-hydraulics	Four equation, drift-flux model in vessel; Two equations for acoustic effects in steam lines; Integral formulation of momentum equation; Boron mass conservation equation; Constitutive relationships for non-equilibrium vapour generation, heat transfer, two-phase pressure loss multipliers; Full core simulation with all fuel channels modelled as parallel channels; Conversion of PDEs into ODEs solved with Euler explicit numerical scheme in core; Runge-Kutta adopted in steam lines; Higher order explicit integration scheme.
Neutronics	3D neutron kinetics based on the PRESTO 11/2 group nodal method; Six delayed neutron groups; Implicit predictor-corrector time integration technique for neutronics and fuel thermodynamics; Full or reduced core symmetry; Nuclear cross-section data parametrized in history (burnup, void history, Xenon) and instantaneous parameters (density, fuel and coolant temperatures, control state, Boron); Boron tracking; Decay heat model ANS Standard 5.1.

TABLE B.17. CHARACTERISTICS OF RAMONA-4B CODE.

Code	**RAMONA-4B**
Property/ Developer	• NRC and BNL, USA
Capabilities	• 1D nonhomogeneous, nonequilibrium two-phase flow in vessel • 3D neutron kinetics for the core • Fixed nodal representation of vessel, 200 core channels, 55 nodes in vessel • Dynamic ex-vessel models for balance of plant, suppression pool, wetwell, drywell and control systems • Dynamic models for SBWR-specific components • Runs on workstations and PCs
Thermal-hydraulics	Four equations drift flux model with nonequilibrium phase change; Loop momentum balance and mixture volumetric flux divergence equations; Up to 200 hydraulic channels with 24 nodes; Nonequilibrium boiling, flashing and condensing in the vessel, equilibrium phase change in containment for relative humidity; Detailed boron transport model with boron mixing efficiency; Polytropic steam flow through turbines (power and FW pump drive, with speed-dependent isentropic turbine efficiency).
Neutronics	3D neutron kinetics based on 1½ group diffusion theory with six delayed neutron groups; Reactivity feedback for Doppler, moderator temperature, void, scram, control rod motion and boron; Decay heat mode consistent with ANS Standard 5.1; Direct gamma heating of liquid phase; Detailed boron tracking in core hydraulic channels.

TABLE B.18. CHARACTERISTICS OF RELAP5 CODE.

Code	**RELAP5/MOD2-RELAP5/MOD3**
Property/ Developer	• Idaho National Engineering Laboratory, USA
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Point neutron kinetics • Detailed noding of the whole plant including BoP
Thermal-hydraulics	1D separated flow model (six equations; complete thermal and mechanical non-equilibrium between the phases); Physical laws selected on the basis of flow regime maps; Semi-implicit and nearly-implicit numerical methods
Neutronics	Point kinetics with six delayed neutron groups; Decay power model; Feedback mechanisms selected by input parameters

TABLE B.19. CHARACTERISTICS OF RETRAN-3D CODE.

Code	**RETRAN-3D**
Property/ Developer	• EPRI, USA
Capabilities	• 3D, 1D parallel channel thermal-hydraulic simulation of the core • 3D, 1D and point neutron kinetics • Detailed noding of the whole plant
Thermal-hydraulics	1D nonequilibrium model with slip; 1D homogeneous equilibrium model with slip; Multi-dimensional vector momentum; Algebraic and dynamic slip correlations; Nodal discretization of the plant; Nondiffusive solution method; General purpose constitutive relationships.
Neutronics	3D, 1D, and point kinetics with six delayed neutron groups; Decay power model; Feedback mechanisms selected by input parameters; 3D and 1D kinetics void and Doppler feedback determined from cross sections.

TABLE B.20. CHARACTERISTICS OF SABRE CODE.

Code	**SABRE**
Property/ Developer	• Pennsylvania Power & Light, USA
Capabilities	• 1D parallel channel thermal-hydraulic simulation of core • Point neutron kinetics • Boron transport • Reactor vessel, steam line, and primary containment modeling
Thermal-hydraulics	Three equation equilibrium model; Locally incompressible flow (no acoustic phenomena); Ohkawa-Lahey void model; Cocurrent and countercurrent flows; Conversion of PDEs into ODEs by control volume formulation; Mixture of implicit and explicit first-order temporal integration
Neutronics	Point neutron kinetics with six delayed neutron groups; Axial power shape varies with time; Reactivity coefficients vary with time and axial position; Nuclear kinetics library based on SIMULATE/SIMTRAN used to compute variations in axial power shape and reactivity coefficients.

TABLE B.21. CHARACTERISTICS OF SIMULATE-3K CODE.

Code	**SIMULATE-3K Code (Version 2.0)**
Property/ Developer	• Studsvik, Sweden-USA
Capabilities	• 3D neutronic core representation • Channel-by-channel thermal-hydraulic representation of the core • Ex-core system simulation
Thermal-hydraulics	Three-equation thermal-hydraulic model using the SETS method with one hydraulic channel per fuel assembly; BWR channel pressure drop balancing; Water rod and bypass flow calculations; Boron advection and maps for modeling asymmetrical loop flows and temperatures (for PWR applications)
Neutronics	Full 3D two-group advanced nodal neutronics model with frequency transform time-integration for accuracy with large time steps, Full SIMULATE-3 cross section libraries (with dependencies on fuel and moderator temperatures, void, control rods, boron, exposure, enrichments, etc.); Neutronic feedback of intra-assembly burnup gradients and spectral histories; Axially zoned control rod models and control rod cusping model; Automatic geometry expansions to full or fractional cores;Automatic time step selection;

TABLE B.22. CHARACTERISTICS OF SPDA (NOW EUREKA-RELAP5) CODE.

Code	**SPDA (now EUREKA-RELAP5)**
Property/ Developer	• Japan Institute of Nuclear Safety (JINS), Japan
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Multigroup 3D representation of the core • Detailed noding of the whole plant • Optional use of coupled of EUREKA/SPACE for neutronics and RELAP5/MOD1 for thermal-hydraulics, or of each code as a separate program
Thermal-hydraulics	1D, 2 phase flow model (five equations; partial thermal, full mechanical non-equilibrium) or 1D flow model (saturated vapour and drift flux); Maximum number of core nodes = (X x Y) 100; Physical laws selected on the basis of flow regime maps; Pre-CHF and post-CHF heat transfer coefficients; Semi-implicit and nearly-implicit numerical methods.
Neutronics	Multigroup time-dependent diffusion in 3D geometry combined with point kinetics; Six delayed neutron groups; Quasi-static solution method; Maximum 2000 material regions; Up to 50 x 50 x 50 space points.

TABLE B.23. CHARACTERISTICS OF STANDY CODE.

Code	**STANDY**
Property/ Developer	• TEPCo, Toshiba Corp. and Hitachi Ltd., Japan, for the coupled multiregion version • TEPCo and Hitachi Ltd., Japan, for the full 3D version
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Element by element neutronic 3D representation of the core • Ex-core component modeling
Thermal-hydraulics	1D separated flow model with 3 equations; Up to 26 axial nodes including 2 nodes for unheated region; Uniform core pressure assumed for property calculation; Compressibility of liquid and vapour neglected; Subcooled boiling considered according to Levy's profile fit; Drift flux according to Zuber-Findlay-Dix; Martinelli-Nelson multiplier for distributed losses and homogeneous flow multiplier for local losses; Iterative scheme comprising: • solution of continuity and energy eqs. for given channel flow • solution of momentum equation for a given core Δp • use of the obtained results to modify core boundary conditions through ex-core models
Neutronics	<u>*Coupled multiregion version:*</u> Up to 20 radial regions composed of arbitrary numbers of fuel bundles (one to several fuel bundles may be included); Up to 24 axial regions; Neutronic coupling between nodes by effective transport kernels; Nuclear constants as a function of coolant density and fuel temperature supplied by a steady-state core simulator <u>*Fully 3D version*</u>*:* Modified one-group time-dependent diffusion in 3D geometry; Six delayed neutron groups; Nuclear constants from a three group model as a function of coolant density and fuel temperature.

TABLE B.24. CHARACTERISTICS OF TOSDYN-2 CODE.

Code	**TOSDYN-2**
Property/ Developer	Toshiba Corp., Japan
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • Element by element neutronic 3D representation of the core • Ex-core component modeling
Thermal-hydraulics	Thermal non-equilibrium, drift-flux model with five equations; 1D parallel channel core simulation; Optional design correlations for drift-flux, heat transfer, etc.; Predictor-corrector solution scheme for continuity and energy; Iterative procedure to achieve the same Δp across the channels, Subcooled boiling treatment.
Neutronics	Modified one-group time-dependent diffusion in 3D geometry; Six delayed neutron groups; Finite difference numerical discretization; One radial node and 24 axial nodes per fuel bundle; Coupling with few thermal-hydraulic channels and fuel structures; Consistency with 3D steady core simulator, used for initialization.

TABLE B.25. CHARACTERISTICS OF TRAB CODE.

Code	**TRAB**
Property/ Developer	Valtion Teknillinen Tutkimuskeskus, Finland
Capabilities	• 1D thermal-hydraulic simulation (at most 3 parallel channels) • 1D neutronics simulation • Ex-core component modeling
Thermal-hydraulics	Four euqation drift-flux model for parallel channels; Two equations for acoustic effects in steam lines; Boron mass conservation equation; Discretization in conservation form; space and time dependent discretization parameter to minimize truncation errors (diffusion).
Neutronics	Two-group diffusion in 1D geometry with radial shape function; Six delayed neutron groups; Two or three radial subregions; 21 to 41 axial nodes; Nuclear constants consistent with VTT nuclear data libraries; Boron tracking; Deacy heat model ANS standard 5.1.

TABLE B.26. CHARACTERISTICS OF TRAC CODE.

Code	**TRAC-BF1 and TRACG**
Property/ Developer	• Idaho National Engineering Laboratory, USA (TRAC-BF1) • General Electric, USA ; Toshiba, Japan (TRACG)
Capabilities	• 1D parallel channel thermal-hydraulic simulation of the core • 3D thermal-hydraulic simulation of the vessel • Element by element neutronic 3D representation of the core • Detailed description of reactor, piping and BOP • Used to determine the setpoints for scram of Oscillation Power Range Monitor (OPRM)
Thermal-hydraulics	Best Estimate thermal-hydraulics with 3D capabilities; Fully non equilibrium two-fluid model with six balance equations; Physical laws applied consistently with flow regime maps; Numerical Method: Stability Enhancing Two-Step (SETS); GE proprietary modifications included in TRACG.
Neutronics	*TRAC-BF1:* Point or 1D neutron kinetics; Six delayed neutron groups *TRACG:* Modified one-group time-dependent diffusion in 3D geometry; Six delayed neutron groups; Equations derived from a three-group model; Quasi-static method for dynamic calculations; One radial node and 24-25 axial nodes per fuel bundle; Coupling with few thermal-hydraulic channels and fuel structures; Consistency with 3D steady core simulator, used for initialization.