The Abdus Salam International Centre for Theoretical Physics

United Nations Educational, Scientific and Cultural Organization

IAEA International Atomic Energy Agency

2152-11

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

17 - 21 May 2010

NATURAL CIRCULATION SITUATIONS RELEVANT TO NUCLEAR POWER PLANTS

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KEY WORDS

Natural Circulation, Water Cooled Reactors, Application of Computational Tools

1. INTRODUCTION

Natural Circulation (NC) is an important mechanism in several industrial systems and the knowledge of its behaviour is of interest to nuclear reactor design, operation and safety. Within the nuclear technology domain, this is especially true for new reactor concepts that largely exploit the gravity forces for the heat removal capability. Natural circulation in nuclear systems occurs due to the presence of the heat source (typically the core) and the heat sink (typically the steam generators in PWR, but also the water pools and the vessel down-comer region in BWR). In a gravity environment, with the heat source located at a lower elevation than the heat sink, driving forces for the fluid motion occur that generate flow-rate suitable for removing nuclear fission power. At present, the NC core power removal capability is only exploited for transient situations, basically to demonstrate the inherent safety features of the nuclear plants.

The present support documentation for a lecture held at the IAEA Course on Natural Circulation in Water-Cooled Nuclear Power Plants (International Centre for Theoretical Physics, ICTP, Trieste, Italy), constitutes one part of a module of six lectures (label <T> in the list of references), see the references D'Auria et al., from 2007a to 2007d.

The present lecture deals with the performance of nuclear reactors in case of NC, following basilar information about the NC in thermal-hydraulic systems given in the mentioned module as well as in a number of other lectures of the Course. This is achieved by describing the results from the application of qualified computational tools in some case supported by experimental data.

Six aspects relevant to natural circulation in NPP or to the modeling of NC are discussed:

a) Flow reversal in U-tubes of steam generators.
b) Characterization of siphon condensation in pressurized water cooled systems.
c) Maximum power removable by NC in PWR.
d) Boron dilution in PWR and NC restart.
e) NC in BWR.
f) NC data from BWR and RBMK relevant to the Natural Circulation Flow Map.

For natural circulation in an inverted U-tube steam generator, item a), it was found that the flow in the parallel U-tubes can be unstable, De Santi et al., 1986 and Sanders, 1988. This is at the origin of important needs in modeling of the steam generators tubes by system thermal-hydraulic codes.

The siphon condensation, item b), has been classified as one of the NC flow regimes in the lecture by D'Auria et al., 2007d. In the present lecture the same NC regime is characterized from a quantitative point of view considering the information from D'Auria & Galassi 1990 and 1990a.

The normal operating condition in PWR is based upon the operation of pumps. However, NC is sufficient to remove part of most of the fission thermal power depending whether single phase or two-phase conditions are tolerated in the core. This subject is discussed under item c) based on the papers by D'Auria et al. 1997 and D'Auria & Frogheri 2002.

The safety relevance of boron dilution in case of small break LOCA (Loss of Coolant Accident) conditions in PWR was stated already in the nineties, e.g. Hyvarinen 1993. Boiling and condensing in steam generators causes the accumulation of boron diluted liquid in the loop seal of the primary loop, item d), Umminger et al. 2002, Reventos et al. 2007. The diluted boron plug may reach the core once natural circulation is restarted, creating the potential for fission reactivity excursion, see D'Auria & Galassi 2001 (see also Giannotti et al. 2006 and Mull et al. 2007).

Boiling Water Reactors (BWR) are characterized by larger vessel related to PWR, including higher elevation and larger thickness for the down-comer. The presence of coolant boiling in the core region (riser) implies the availability of higher driving forces. Thus the NC mechanism can be and actually is efficiently used to remove higher core power than in the case of PWR, item e). The operating power-to-flow map of the BWR NPP, see also D'Auria et al. 2007c, includes a part where NC occurs. The objective here is to characterize the NC part of the BWR operating map, Ambrosini et al. 2003.

The derivation of the Natural Circulation Flow Map (NCFM) for PWR has been presented in the lecture by D'Auria et al. 2007d, where examples of application to pressurized systems are given. Within the present framework, the use of the NCFM is extended to nuclear boiling systems, including BWR and RBMK, D'Auria et al. 2008.

The discussion in the following chapters 2. to 8. deals with the topics a) to e), respectively, and is based upon the listed references where additional details can be found.

2. FLOW REVERSAL IN U-TUBES OF STEAM GENERATORS

During single phase natural circulation in the LOBI facility, see D'Auria et al. 2007d, non uniform mass flow distribution among the U-Tubes (8 or 24 full length U-Tubes constitute the steam generators for the 'broken' or the 'intact' loop, respectively) in the primary system was observed, De Santi et al. 2006. There were tubes with steady flow of liquid, while in at least one tube the liquid was completely stalled. The experimental evidence of the tubes performance was based on the signal of thermocouples installed along the tube length. Tubes with flow showed small temperature differences along the fluid flow direction and tubes in stalled conditions (nearly zero flow) showed temperatures in equilibrium with the secondary side.

The phenomenon was previously observed in the Semiscale facility with non uniform flow occurred in several tubes, Loomis et al. 1981. Later on, Kukita et al. studied the phenomenon occurring in the Japanese facility LSTF and recently, Spadoni et al. mentioned the problem (see below) connected with flow reversal in steam generator U-Tubes of the PKL-III facility.

The problem of technological interest associated with the occurrence of flow reversal during natural circulation in PWR type NPP can be summarized in four paragraphs as follows:

1) The thermal-hydraulic phenomena occurring in case of the NC scenario are expected to occur in a number of transients within the Design Basis Accident conditions. These include a number of small break LOCA, the Loss of Feed-Water and the Station Blackout transients.

2) Within the licensing process, but also for safety analyses and design optimisation purposes, a suitable input deck of any NPP must be developed for the application of system thermal-hydraulic codes (e.g. analysis of transient scenario at the item above).

3) The NPP model for system thermal-hydraulic codes typically include a number of loops as in the plant and each loop includes one steam generator equipped with a bundle constituted by around 4000 U-tubes. The entire U-Tube bundle is simulated owing to a number of reasons, (one important reasons being the available computer memory) by one equivalent thermal-hydraulic zone typically sub-divided into 10 – 50 nodes along the unique flow-path.

4) It is evident that a standard input model suitable for system thermal-hydraulic code is not capable of simulating the different behaviours of different U-tubes or groups of U-Tubes. Thus, parallel thermal-hydraulic zones shall be built to simulate the U-tubes bundle with such a code (in this case, however, some input quantities like the number of tubes to be lumped and the local pressure drop coefficient of each lumped tube group with the inlet or outlet plenum are un-defined and code-user assumptions are typically needed, thus 'forcing' the different predicted behaviour). The consequence of the modelling deficiency occurring when only one group of tubes is simulated is the overestimation of the net flow through the loops and therefore the overestimation of the core flow-rate in case of natural circulation and of the NC core cooling potential.

The inadequacy in predicting NC in both single phase and two-phase conditions caused by flow reversal, superimposes with the inadequacy in the modelling of local pressure drops at low values of Reynolds number as discussed by D'Auria & Galassi 1992.

3. CHARACTERIZATION OF SIPHON CONDENSATION

Oscillations were measured and calculated for several thousands seconds into the transient during one NC experiment performed in LOBI facility (test A2-77a) as also discussed by D'Auria et al. 2007d. The oscillations were observed into what was preliminarily called 'transition period' between two phase NC and reflux condensing mode (e.g. see D'Auria & Galassi 1990a) and later on called 'siphon condensation NC regime (or mode)'. A picture of the oscillations can be found in Fig. 1. The following can be noted:

- The oscillation period is consistent with the timing of filling of the ascending side of the U-tubes caused by simultaneous occurrence of condensation on the tube walls and Counter-Current Flow Limitation (CCFL) at the tube inlet.
- Once the ascending side of the U-tube is full of liquid the siphon effect occurs with sudden draining of the tube and restart of the flow-rate.
- The oscillations are reflected all around the loop and shown by the measured time trends of fluid velocities, pressure drops and densities in both the loops of the facility (see also Fig. 1).
- The adopted code (Relap5) was able to catch the main parameters of the oscillations (e.g. Figs 1c and 1d) and was of help in characterizing the overall system behaviour.
- The same type of oscillation was observed in other experimental facilities simulating the behaviour of PWR, e.g. Kukita et al. 1988.

A further investigation, primarily using the Relap5 system code, by D'Auria & Galassi 1990, brought to two main outcomes:

1) Groups of U-tubes may exhibit different behaviour at the same time with boundary conditions assigned in the inlet and the outlet plenum of the steam generators. Namely, some tubes must be in stalled conditions in order to have CCFL in others. Furthermore, the height of tubes has little influence on the overall evolution of the phenomenon (i.e. several tubes of the same height still exhibit the siphon-condensation mode). Rather, two-dimensional fluid dynamics inside the steam generator plenum differentiate the U-tube behaviour. Finally the CCFL front is likely to advance in some U-Tubes. The sketch in Fig. 2, was used to depict the siphon-condensation phenomenon.

2) The siphon condensation phenomenon is expected to occur in NPP conditions.

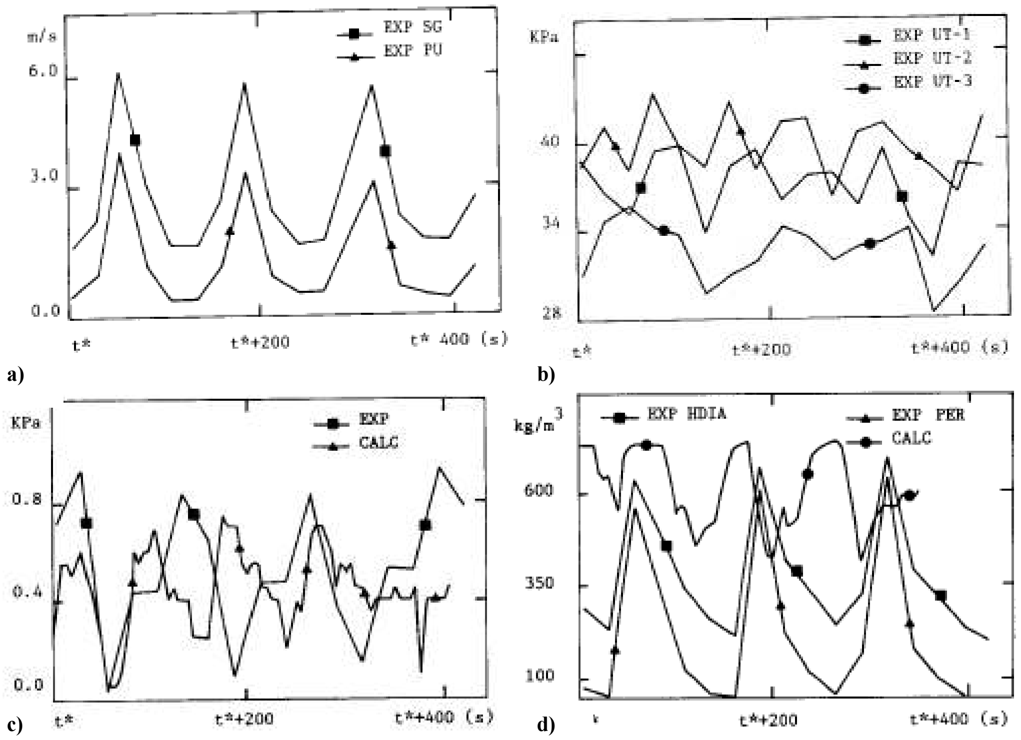


Fig. 1 – LOBI NC test A2-77a, measured and calculated data: a) experimental signal for fluid velocity of intact loop at steam generator (SG) and pump (PU) inlet; b) experimental signal of pressure drop in the ascending leg of three different U-Tubes of the intact loop steam generator; c) comparison between measured and calculated values of pressure drop across the broken loop steam generator; d) comparison between measured and calculated values of fluid densities (by 'peripheral' and 'diameter' gamma-densitometer, 'PER' and 'HDIA' respectively) in the cold leg of the broken loop.

4. MAXIMUM POWER REMOVABLE BY NC IN PWR

The objective for this study, D'Auria et al. 1997, was to gather an estimation of the maximum thermal power removable by NC in PWR systems. This may be relevant for the existing reactor technology in the extremely unlikely event of main circulation pump failure and simultaneous failure of scram and in the design of new reactors fully based upon natural circulation. No care is given in the analysis to the thermal-hydraulic neutron kinetics interaction and, therefore, to the actual possibility of generating the fission power needed to sustain the NC.

The analysis is carried out into two parts by the use of the Realp5 system thermal-hydraulic code: the former related to Integral Test facilities (ITF) to use qualified nodalizations and to derive a qualitative idea of the system performance, the latter related to a PWR nodalization.

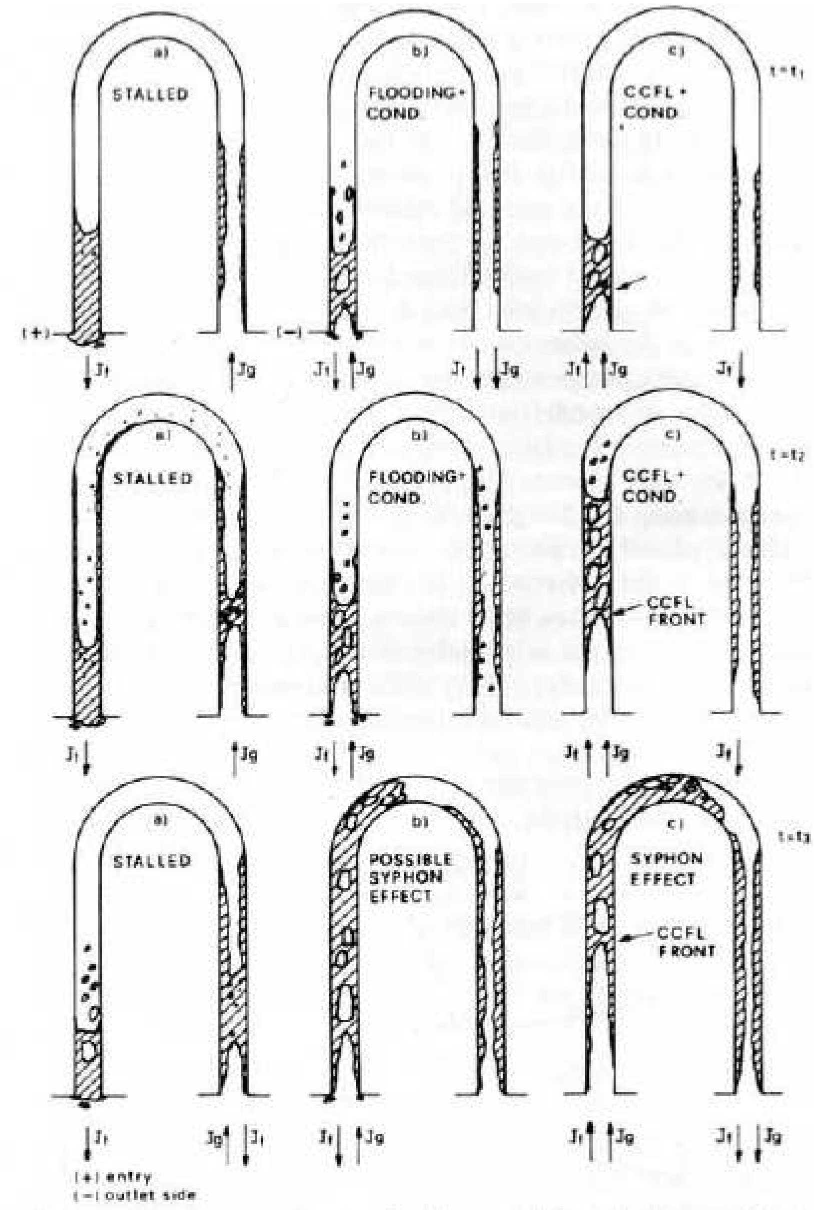


Fig. 2 – Characterization of the siphon-condensation NC regime: at each time t_1,t_2 and t_3, groups of different numbers of U-tubes are in the configurations a), b) and c).

In all planned code runs, the primary system pressure and the steam generator relevant parameter values (e.g. pressure, level and feed-water temperature) are kept constant at the nominal values or consistently assigned: for instance, the feed-water flow-rate is set proportional to the core power. The main circulation pumps are set at zero speed and the locked rotor hydraulic resistance of the impeller is taken into account. The core power level is assumed as the independent variable and is varied till obtaining one of more of the following thresholds:

➔ Saturated void appearance at core outlet (this is assumed to occur when void fraction achieved the value 0.1 at the uppermost hydraulic node of the core) as the boundary for single phase NC.

➔ Occurrence of dry-out in the highest linear power rods of the core. A number of thermal-hydraulic parameters are associated with this occurrence like core power, mass inventory in the loop and ratio 'core mass flow-rate / core power' (G/P in Kg/Mw-s).

➔ Rod surface temperature over-passing the 600 °C (or 873 K). This threshold value is not detected in the performed analyses, but can be relevant for design purposes.

The key results of the first part of the study, related to ITF, are summarized in Tab. 1 and the following notes apply:

- A uniform increase in NC flow-rate is predicted when core power is increases, until core power achieves a value of around 40% of the nominal value. Further increases of core power do not cause proportional increases in core flow.
- Oscillatory flows are calculated for core power larger than 40% in Bethsy and Lobi ITF.
- The primary mass inventory decrease occurs via the pressurizer relief valve that is assumed to open and to close in order to maintain constant the system pressure.
- PWR cores, in their current-actual geometric and material configurations are expected to have the capability to operate in NC conditions, i.e. to have sufficient amount of moderator-coolant in the core to sustain the fission reaction, when the power has values up to about 15% the nominal value.
- The largest facilities are designed to operate at low core power (ITF design finalized to the simulation of small break LOCA): this may explain the small value of % core power at which dry-out occurs in Lstf. Neglecting the Lstf case, up to 70% core power can be removed by NC before experiencing dry-out. This can be assumed as the thermal-hydraulic limit for system (that should not any more be classified as PWR) operation in NC.

TABLE 1. – REMOVABLE POWER BY NATURAL CIRCULATION IN ITF.

ITF	Core power when void achieves 0.1 at the upper core level (°)	Core power when dry-out occurs (°)	Void at the upper core level when dry-out occurs	Primary system mass inventory at dry-out (°)	G/P at dry-out (Kg/MWs)	RM/V at dry-out (Kg/m3)
Bethsy	15	70	0.8	69	1.12	475
Lobi	20	70	0.7	80	1.23	570
Lstf	10	30	0.9	62	1.87	480
Spes	15	50	0.6	75	1.29	528

(°) % of the nominal operational value

The second part of the study was carried out by utilizing a qualified PWR NPP nodalization suitable for the Relap5 code. A summary of performed code runs, including significant results, is reported in Tab. 2.

All the reported data relate to conditions where core power equals SG removed power. This is also valid when dry-out occurs: in those situations, "small" (i.e. of the order of 10 K) excursions of rod surface temperature are calculated.

The NC core flow-rate is largely affected by the void content in the loop. The mass inventory in the primary loop and the upper plenum void values, reported in Tab. 2, give an idea of the void content in the loop. At limit values for core flow-rate, i.e. close to the dry-out condition, non linear relationship is calculated between core flow-rate and core power. A complex relationship also links core flow-rate and the steam generator conditions including pressure and feed-water temperature.

A number of code runs have been performed to demonstrate the possibility of operating the NPP at 100% core power in NC without dry-out occurrence. Boundary conditions in primary and secondary system have been varied to this aim, e.g. cases 21, 23 and 25 in Tab. 2.

TABLE 2 – REMOVABLE POWER BY NATURAL CIRCULATION IN PWR.

No	ID	P MW/%	G (Kg/s)/%	SG PRE MPa	RM KgE5/%	PS PRE MPa	UP T/Tsat K	UP Void -	G/P Kg/sMW	RM/V Kg/m3
1#	KK01	1876/100	9037/100	6.1	1.08/100	15.6	598/618	0.	4.82	647
2^	KK01	56/3.0	520/5.8	8.1*	1.08/100	13.6	577/608	0.	9.28	647
3	KK01	376/20.	930/10.3	6.0*	1.08/100	15.4	615/617	0.	2.47	647
4	KN03	469/25.	1016/11.2	6.0*	1.07/99.1	16.2	620/620	0.10	2.17	641
5	KN04	563/30.	1140/12.6	6.0*	1.01/94.0	16.2	620/620	0.21	2.02	605
6	KN05	938/50.	1370/15.1	6.0*	0.92/85.0	16.2	620/620	0.47	1.46	550
7	KN07	1032/55.	1396/15.4	6.0*	0.90/83.3	16.2	620/620	0.48	1.35	539
8	KN08	1126/60.	1428/15.8	6.0*	0.89/82.9	16.2	620/620	0.49	1.27	536
9	KN09	1219/65.	1450/16.0	6.0*	0.88/82.0	16.2	620/620	0.51	1.19	529
10§	KN10	1313/70.	1490/16.4	6.0*	0.87/80.8	16.2	620/620	0.62	1.13	523
11	KL10	1032/55.	1396/15.4	3.5*	0.99/91.4	16.2	620/620	0.44	1.35	592
12	KL10	1313/70.	1650/18.3	3.5*	0.95/88.2	16.2	620/620	0.49	1.26	571
13§	KL12	1500/80	1492/16.5	3.5*	0.91/84.6	16.2	620/620	0.60	0.99	547
14§	KL11	1688/90.	1523/16.8	3.5*	0.87/80.4	16.2	620/620	0.77	0.90	520
15	LL11	1032/55.	1365/15.1	3.5**	1.01/93.9	16.2	620/620	0.31	1.32	608
16§	LL11	1688/90.	1525/16.9	3.5**	0.93/86.3	16.2	620/620	0.57	0.90	556
17	LL12	1500/80.	1380/15.3	3.5**	0.96/88.8	16.2	620/620	0.49	0.92	575
18	LL13	1032/55.	1300/14.4	2.5**	1.04/96.3	16.2	620/620	0.20	1.26	623
19	LL13	1500/80.	1750/19.4	2.5**	1.00/92.3	16.2	620/620	0.48	1.17	597
20	LL14	1688/90.	1460/16.2	2.5**	0.97/89.4	16.2	620/620	0.50	0.87	578
21§	LL15	1876/100.	1587/17.5	2.5**	0.94/86.6	16.2	620/620	0.63	0.85	560
22	HL15	1032/55.	1290/14.3	2.5**	1.09/101.	18.5	631/633	0.01	1.25	652
23§	HL15	1876/100.	1630/18.1	2.5**	0.97/89.3	18.5	633/633	0.58	0.87	578
24	HL16	1032/55.	1295/14.3	2.5+	1.09/101.	18.5	594/633	0.	1.25	652
25§	HL16	1876/100.	1630/18.1	2.5+	0.99/91.4	18.5	633/633	0.50	0.87	590

Nomenclature

ID	**Calculation identification**	**§**	**Dryout occurrence**
G	**Core flowrate**	**#**	**Nominal working conditions for the system**
P	**Core Power**	**^**	**Reference NC result**
PRE	**Pressure**	*****	**Feedwater temperature same as in nominal condition**
PS	**Primary System**	******	**Feedwater temperature set at 363 K**
RM	**Mass Inventory in PS**	**+**	**Feedwater temperature set at 333 K**
T	**Fluid Temperature**	**+**	**Feedwater flowrate set at 1.3 times the equilibrium value**
Tsat	**Saturation temperature**	**UP**	**Upper Plenum**
Void	**Void fraction**		

Additional results from the study are given in Fig. 3 and related key comments (Figs. 3a to 3d and Tab. 2) are:

- Single phase natural circulation can be maintained up to about 20% core power, thus confirming the results related to ITF (Tab. 1).
- Two-phase NC allows the removal of up to about 70% core power assuming nominal system conditions, again confirming the results related to ITF.
- Lowering the steam generator pressure and increasing primary system pressure bring to increases in the NC thermal power removal capabilities. More than 90% core power can be removed in NC with SG pressure as low as 2.5 MPa (see also Fig. 3d).
- Dry-out occurrences are undesirable. However temperature excursions are limited (as already mentioned) and do not affect the 'stable and steady' NC scenario.

- ❖ The NC flow map, see D'Auria et al. 2007d, is derived for low values of core power (decay heat), but appears useful also for high power values, Fig. 3a. Dry-out can be seen with G/P close to unity (fixed units) or below this threshold, Fig. 3a, see also Fig. 3c. Lower values of G/P at dry-out are experienced at higher core power.
- ❖ NC flow-rate is an increasing function of the core power (as expected, Fig. 3b). However, when dry-out conditions are experienced, the slope of the curve G versus P largely decreases, Fig. 3b.

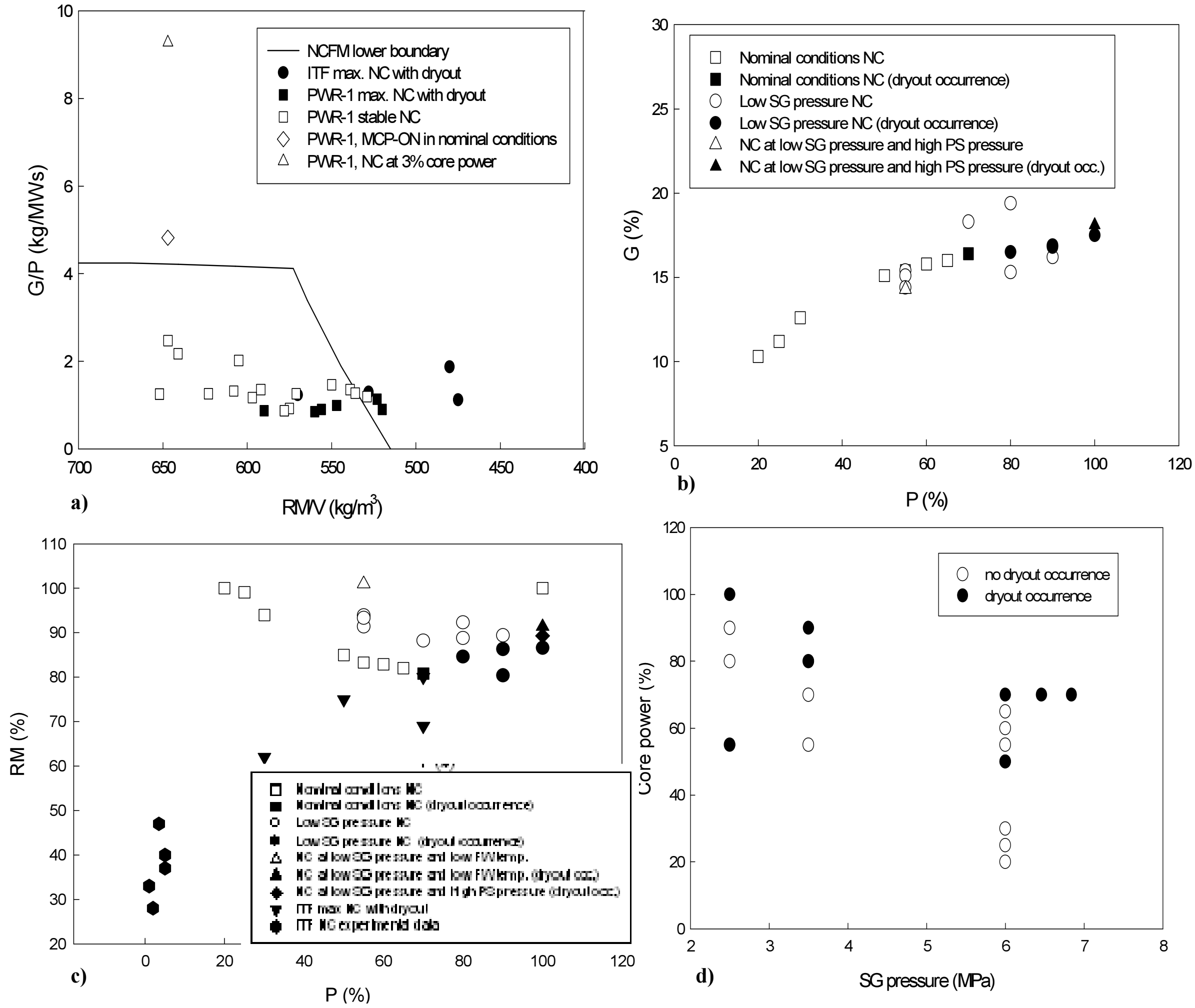


Fig. 3 – Maximum power removable by NC in a PWR (reference is made in each diagram to selected code runs in Tab. 2) : a) consideration of the Natural Circulation Flow Map, see D'Auria et al. 2007d; b) predicted NC mass flow rate versus core power; c) primary system mass inventory versus core power; d) core power and steam generator pressure.

5. BORON DILUTION AND RESTART OF NC

During the course of a Small break LOCA in a PWR it may happen that, during a more or less wide time period (this may be a few tens of seconds as well as several minutes), the primary system coolant inventory remains constant at values in the range (40 ÷ 70) % of the initial (nominal) values. This depends upon the break size and location as well as upon the modalities of actuation of the emergency systems including location, time of actuation and overall flow-rate.

In the case this condition takes place, boiling in the core and condensation in steam generators is expected and the scenario is depicted as two-phase natural circulation including siphon condensation and especially reflux condensation regimes (see D'Auria et al. 2007d). In the case the event occurs at beginning of life with 'high' boron concentration in the primary coolant, boron-diluted plugs may form in the loop seals.

Now, a few tens of seconds are needed to form diluted boron plugs and to fill loop seals with a volume (of 'clean' liquid) larger than the 'free' core volume (i.e. the volume occupied by coolant in the active core region). At the same time, liquid with 'high' concentration of boron is expected in the vessel. In this period the natural circulation flow-rate to the core may be very low (e.g. reflux condensing flow regime) also due to the presence of the break, and boron diluted and highly borated liquid zones may form in the primary coolant loop.

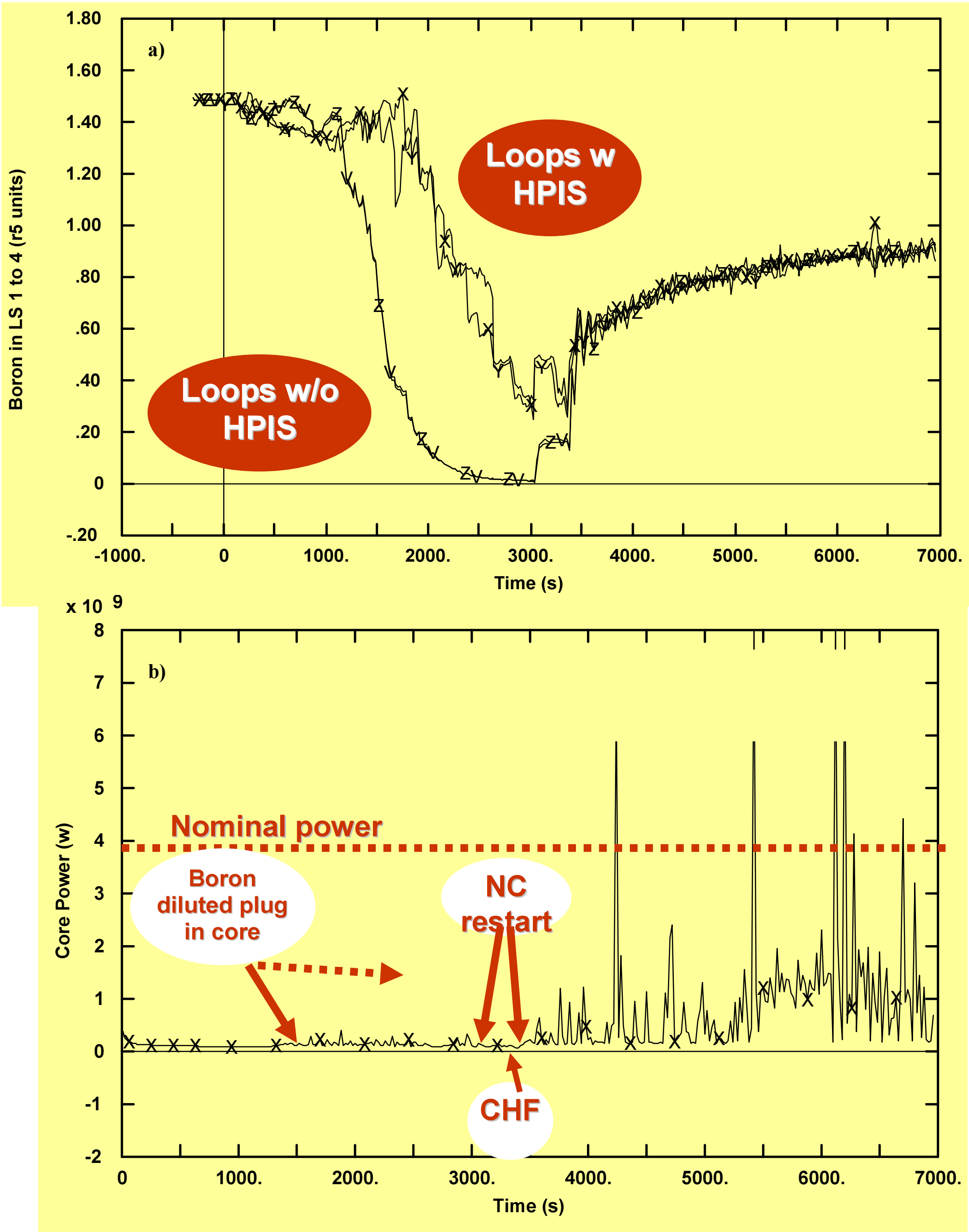


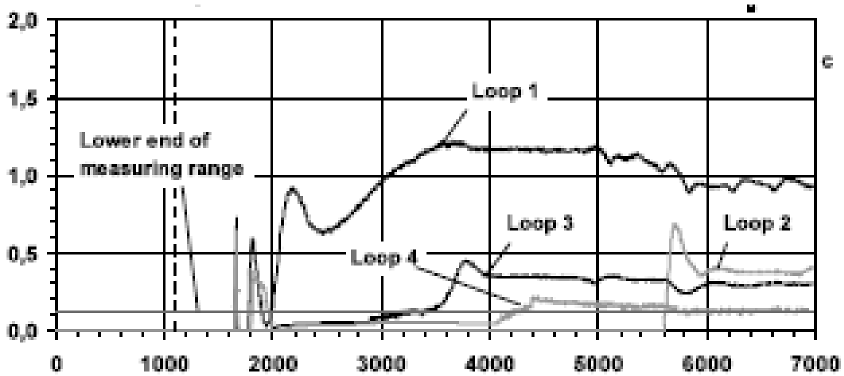
Fig. 4 – PWR NPP calculation of a Small Break LOCA scenario: a) average boron concentration in four loop seals; b) core power.

The recovery of the system unavoidably implies the refill of the primary loop and the restart of natural circulation up to the 'stable long-term situation' of single phase natural circulation. At the time of restart of the natural circulation, the problem may appear: the boron diluted plug from loop seal enter the vessel and displaces (other than mixing) the highly borated water and enters the core creating the potential for reactivity excursion.

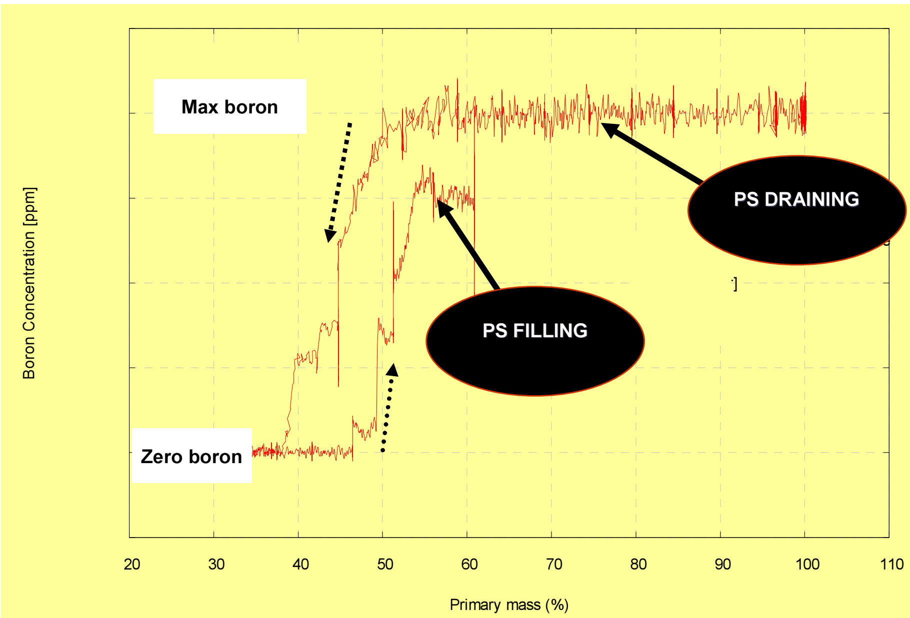
The scenario has been calculated by D'Auria and Galassi 2001 for a PWR and is depicted in Fig. 4.

Arbitrary units are reported in the vertical axis of Fig. 4a). It can be observed that it takes around 15 minutes in the considered case to have the full deboration of the liquid in the loop seals and that the phenomenon is marginally affected by the actuation of the High Pressure Injection Systems (HPIS) inn two of the four loops of the NPP. Furthermore, in the considered case the NC restart occurs at around 3000 s into the transient. The "arrival" of the boron plug into the core causes re-criticality, thus fission power rise and Critical Heat Flux (CHF), as given in Fig. 4b). Complete mixing is assumed in the vessel by the thermal-hydraulic model that is coupled with a point kinetics model.

Results from NC experiments performed in PKL-III facility are shown in Fig. 5, Umminger et al. 2002 (see also Mull et al. 2007). The experiments are used to provide information (as far as possible) upon the envisaged NPP scenario and primarily to qualify the computational tools used to perform NPP calculation.



a)



b)

Fig. 5 – Boron dilution during a NC experiment in the PKL-III ITF: a) flow-rate in each loop (NC restart) following the system refilling; b) boron concentration during the Primary System (PS) draining and filling.

The data in Fig. 5a) show that NC flow-rate does not restart simultaneously in the four loops (thus the transient in the vessel can be milder related to the case when the NC restarts simultaneously in all loops) and the value of the flow-rate is different. The data in Fig. 5b) show measurement of boron concentration in the loop seals a function of mass inventory in the primary system during the cycle of mass draining and filling (hysteresis can be observed) and confirm (from a qualitative point of view) the time trends predicted in Fig. 4b.

6. NATURAL CIRCULATION IN BWR

The operation power-to-flow map of a BWR has already been discussed by D'Auria et al. 2007c. The attention is focused hereafter to the left part of the map where NC parameters are reported.

Chimney type and drum type BWR are distinguished by Abe et al. 1994 in relation to NC characteristics. In the same paper a simplified relationship is proposed to obtain NC flow in BWR. Three diagrams are proposed hereafter, Figs. 6 to 8, to illustrate the NC performance of BWR systems, Ambrosini et al. 1993, Abe et al. 1994 and Lennern 2006.

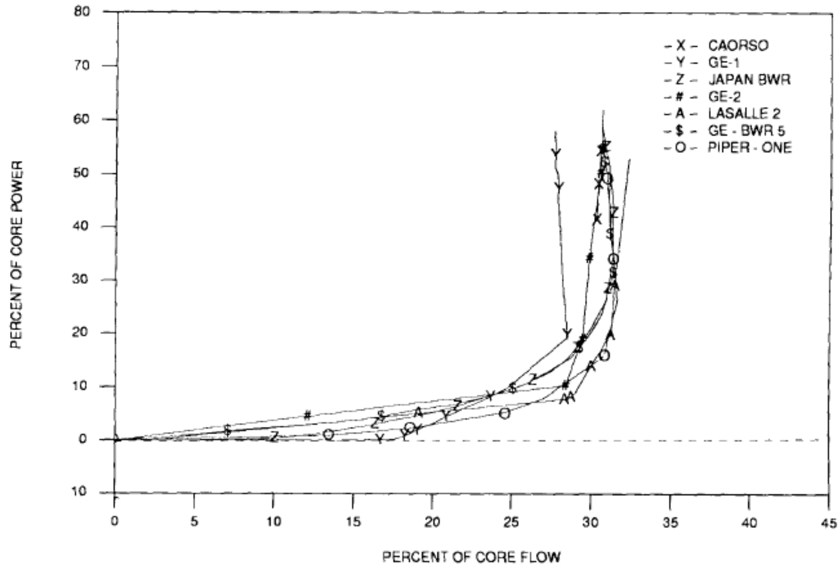


Fig. 6 – NC in BWR systems: portion of the power-to-flow map in currently operating BWR and in the ITF simulator Piper-one, Ambrosini et al. 1993.

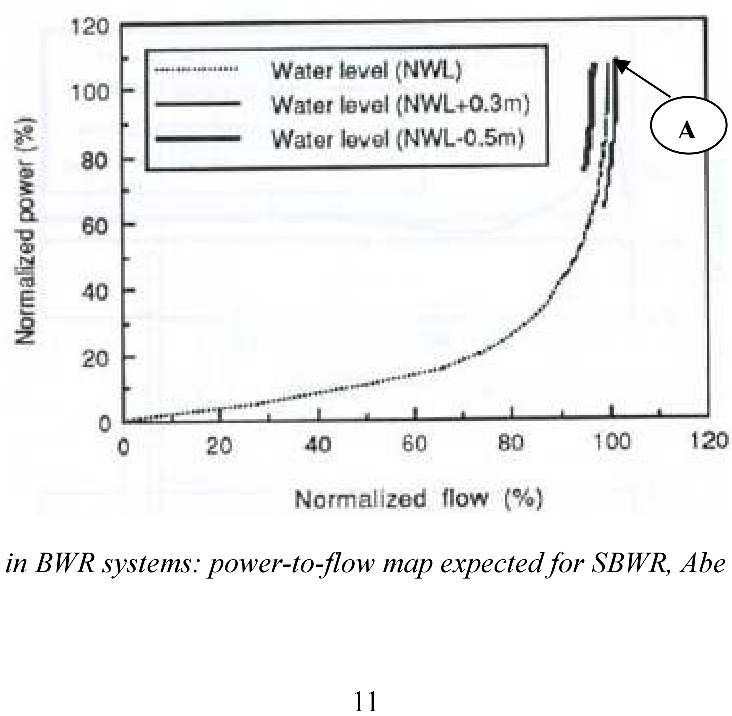


Fig. 7 – NC in BWR systems: power-to-flow map expected for SBWR, Abe et al., 1994.

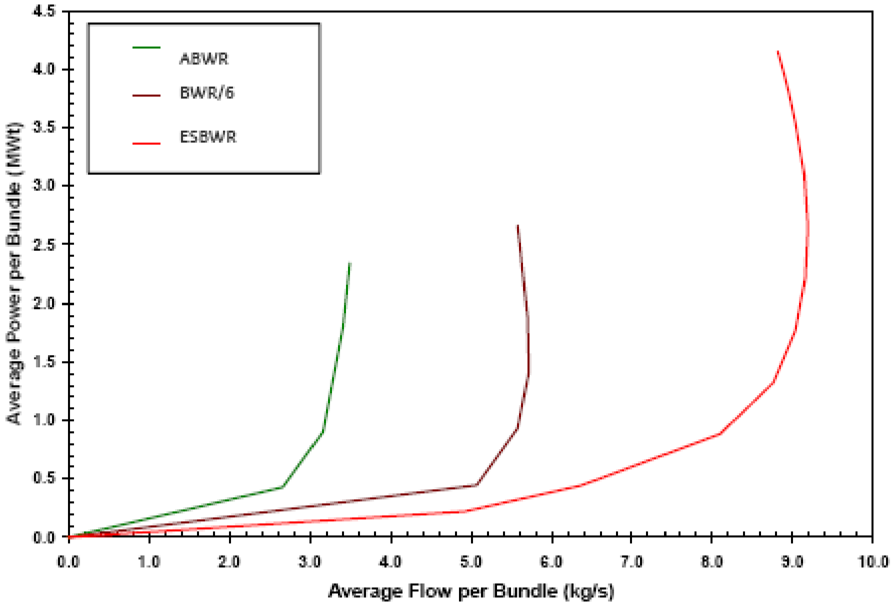


Fig. 8 – NC in BWR systems: comparison of power and flow per fuel bundle in current BWR (BWR/6), ABWR and ESBWR, Lennern 2006.

The first diagram, Fig. 6, shows that various BWR now in operation have very similar natural circulation characteristics. Furthermore, the PIPER-ONE ITF (full height and 1/2000 volume and power scaling factors) has the capability to reproduce the NC characteristic of the prototype (BWR/6) NPP. The removed power is almost independent upon the flow-rate for flow-rate values around 30% of the nominal value: this implies an increasing average core void fraction when power is increased. The NC curve is part of the operating conditions for NPP till power values close to 50% of the nominal value; for greater power values coolant pumps (and forced circulation) are needed. It may be noted that a PWR may operate in NC conditions till power values close to 15% of the nominal value (section 4), while the same limit is around 50% in the case of BWR.

The second diagram, Fig. 7, shows the improved NC capabilities of a SBWR. In this case NC is capable of removing 100% of the generated core power in nominal conditions (better: the SBWR design is such that nominal power is removed by NC). The superposition of diagrams in Figs. 6 and 7 assuming the axes units in % of the quantities for a BWR, shows that the point “A” of Fig. 7 would be located in Fig. 6 (not reported in the diagram) at values of both power and flow-rate equal to 60%.

A direct comparison of power-to-flow, in NC conditions, per individual fuel bundle in different generations of BWR, i.e. BWR/6, ABWR and ESBWR, is given in Fig. 8. The good NC characteristics of ESBWR are achieved by reducing the pressure drop in the riser region of the vessel (including the core) and by increasing the level in the down-comer (i.e. level elevation related to core). It is interesting to note the values of G/P at the top of each curve for ABWR, BWR/6 and ESBWR: these values are around 1.48 for ABWR and 2.15 (kg/Mw s) for both BWR/6 and ESBWR and can be considered for comparison with PWR in the vertical axis of Fig. 3a.

7. THE USE OF THE NC FLOW MAP FOR BWR AND RBMK

The Natural Circulation Flow Map (NCFM) was derived for PWR equipped with U-Tubes steam generators, e.g. D’Auria & Frogheri 2002. Its use was extended to pressurized water cooled reactors equipped with once-through and horizontal steam generators and to CANDU systems, e.g. D’Auria et al. 2007d. Now, having in mind the objective of evaluating the thermal-hydraulic system design of the RBMK systems, e.g. D'Auria et al. 2008, operational points that are characteristic of BWR and RBMK have been reported in the NCFM, Fig. 9.

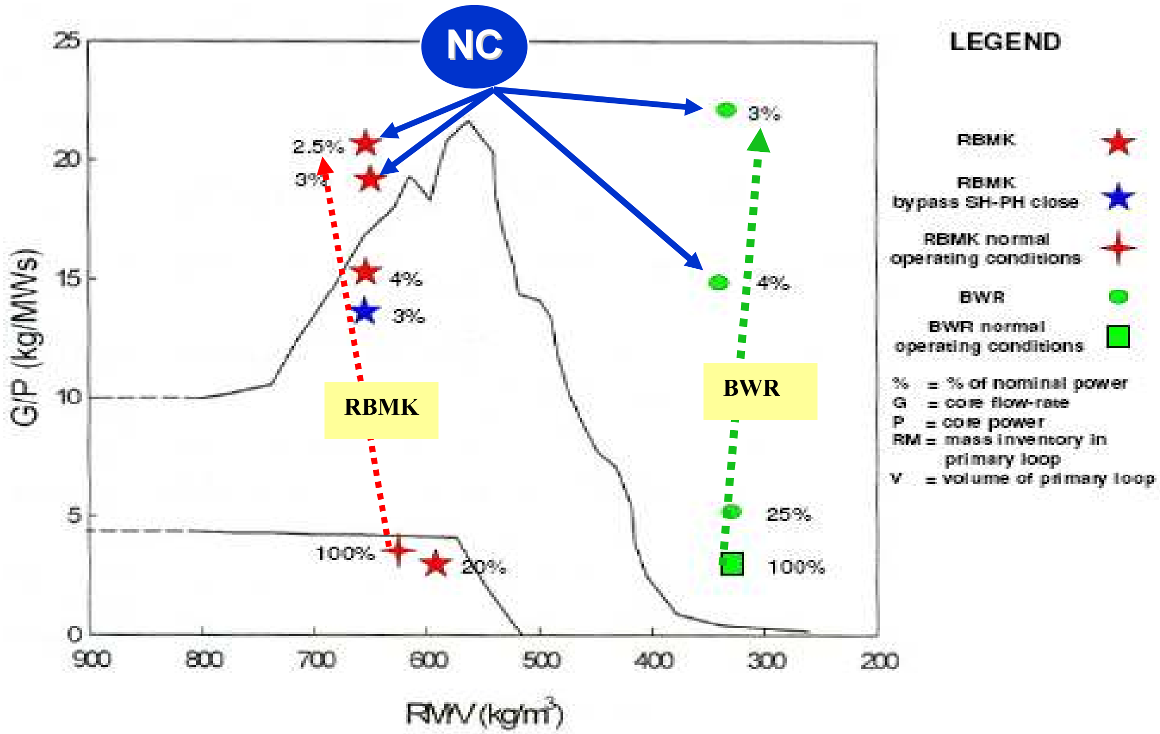


Fig. 9 – Use of NCFM for BWR and RBMK systems.

In the case of both BWR and RBMK, the values on the horizontal axis can be hardly defined, because of the lack of definition of the primary loop (this is a closed loop in case of PWR) and then of the volume (for instance the steam line volume is not irrelevant compared with the vessel free zone or the core volume in the case of RBMK). Therefore the location of working points along the horizontal axis in Fig. 9 is arbitrary and irrelevant for the study.

RBMK and BWR should be considered as drum type and chimney type according to the definition by Abe et al. 1994. The related operational points are reported on the left and the right side of the diagram, respectively, with % values indicating the concerned power level related to the nominal value. All reported data have been obtained by qualified calculations carried out by Relap5. The NCFM has been derived at decay power values therefore the points characterized by values of 4% or less power are directly comparable. The following can be added:

- Nominal operating points appear in the bottom part of the diagram: this is consistent with high power NC in PWR as given in Fig. 3a.
- High power NC (e.g. 20% and 25% points in Fig. 9), are close to full power in terms of G/P: this also applies to the maximum NC flow-rate as mentioned in the last paragraph of chapter 6 above.
- There is an important difference in terms of G/P when the power is reduced from the nominal 100 % down to 4% values and from 4% to 3% values (or 2.5% in case of RBMK).
- RBMK and BWR exhibit very similar NC performance notwithstanding the different system layout and the longer core (low equivalent diameter and higher pressure drop) in case of RBMK. The reason for the equivalent NC performance comes from the elevation of the steam drum and the consequent higher driving head available in the case of RBMK (around 15 m

and less than 5 m for RBMK and BWR, respectively, if one considers the distance between the level in the down-comer region and the top of the active fuel).
- In the case of RBMK two values at 3% core power shall be noted: these are related to the operating of a pump bypass loop that is expected to operate when the primary pumps are tripped.
- At low power values the high NC flow-rate of BWR and RBMK comparable with the peak flow-rate in PWR, shall be noted.

The performed study proved the suitable (state-of-the-art) NC performance of the RBMK systems.

8. CONCLUSIONS

An overview of application of natural circulation to the technology of water cooled nuclear reactor has been given. This should not be considered as an exhaustive analysis, rather as a starting point for the reader for deeper understanding of the nuclear reactor design. The main lessons learned can be summarized as follows:

a) The investigation of NC in water cooled reactors requires properly (specifically) qualified computational tools: this is true in particular for the dependency upon Reynolds number (and upon local void fraction) of the local pressure drop coefficient and for the development of suitable nodalizations (e.g. number of parallel U-tubes in steam generators).
b) System effects cause oscillations in selected quantities. The origin of the oscillations in case of siphon condensation is the system geometry. Those oscillations in PWR are expected during unlikely LOCA type event and should not be compared in terms of safety relevance with the density wave oscillations in BWR (these constitute the topic of a specific lecture, e.g. D'Auria et al. 2007c).
c) NC as a thermal-hydraulic mechanism is suitable to remove large power fraction in PWR, provided the possibility to maintain the fission reaction at low core density conditions.
d) Boiling and condensing phenomena, typical of two-phase NC in a PWR, are at the origin of boron dilution in zones of the primary loop. The diluted boron plugs may reach the core and cause the potential for reactivity excursions.
e) The Natural Circulation Flow Map (derived for PWR) confirms to be an interesting tool to evaluate the NC performance of systems different from PWR.
f) BWR and RBMK have similar NC performance and confirm the superior NC performance of two-phase systems related to single phase systems (within the range of parameters of interest). The adequacy of the thermal-hydraulic design of RBMK is also confirmed from the viewpoint of NC.

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