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USE OF NATURAL CIRCULATION FLOW MAP -UPDATED

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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. In the nuclear technology, this is especially true for new concepts that largely exploit the gravity forces for the heat removal capability. Natural circulation in a PWR (Pressurized Water Reactor) occurs due to the presence of the heat source (core) and the heat sink constituted by the steam generators. In a gravity environment, with core located at a lower elevation than steam generators, driving forces occur that generate flow-rate suitable for removing nuclear fission power. Within the nuclear reactor technology, the capability to remove thermal power by NC is mainly exploited for accident situations, thus demonstrating the inherent safety features of the existing plants (IAEA 2002 and IAEA 2005).

From the thermal-hydraulic point of view, simple and complex loops should be distinguished, with the simplest (unstable) loop proposed by Welander, e.g. see the companion (in the same Course) lecture by D'Auria et al. 2007, and complex loops constituted by natural circulation nuclear systems, e.g. see IAEA 2006.

In the present document reference is made to the natural circulation in complex systems, considering the experimental simulators of PWR and the experience gained in analyzing natural circulation experiments performed in those systems, e.g. D'Auria et al. 1991 and D'Auria et al. 1998. In particular, the NC scenarios occurring at different values of the primary system mass inventory are studied. Data have been gathered and analyzed coming from the experimental facilities (PWR simulators, or Integral Test Facilities, ITF) named Semiscale, Spes, Lobi, Bethsy, Pkl and Lstf.

In the case of PWR design, the NC should be seen as the fundamental mechanism for removing core power following accidents or events that cause unavailability of primary circulation pumps. The NC phenomenon constitutes the basis for the layout of the primary system with core located in the bottom of the vessel and steam generators a few meters above the top of the active fuel. Assuming the availability of steam generator cooling, the NC flow-rate shall be such to ensure the removal of the decay power from the core at various values of the coolant mass inventory in the primary system. Therefore ensuring a 'good' natural circulation performance for the system implies addressing safety issues in different accident scenarios part of the 'design basis envelope' and answering several questions coming from the licensing process of the NPP (Nuclear Power Plant). Owing to this reason, the NC experiments have been performed in all major ITF built and operated so far worldwide.

Both single phase and two-phase natural circulation situations are considered here. The objective is to analyze the experimental data base available from the mentioned ITF and to establish a NC map suitable to evaluate the NC Performance (NCP) in complex nuclear systems, primarily PWR; the NC capabilities of other complex systems, including '*innovative reactors*' can be demonstrated against the NCP map.

The information to built-up the NCP map at assigned pressure of primary system (i.e. around the saturation pressure that characterizes the coolant in nominal conditions), at assigned core power (i.e. around the decay power a few tens of seconds after the scram), includes the time evolutions of the core inlet mass flow-rate and the primary loop coolant mass inventory. The flow-rate and the residual mass, or mass inventory, have been normalized taking into account the primary system volume of each facility and the power level.

Four main flow patterns are characterized in advance and discussed in chapter 2, then the NCP map is derived (chapter 3), and used for establishing the NC characteristics of nuclear systems not at the origin of the map itself (chapter 4).

2. THE NATURAL CIRCULATION FLOW REGIMES IN PWR

Let us consider PWR NPP working at nominal operating conditions for a suitable time period. Let us assume the trip of the pumps and the simultaneous scram with feeding system always available in the secondary side to steam generators (heat sink). In this situation the nuclear reactor core power establishes at decay values (heat source) and the primary system pressure stabilizes at a value close to the saturation pressure corresponding to primary fluid nominal operating conditions, i.e. in the range 7 to 9 MPa. Summing up, the NC system constituted in the primary system along the fluid flow path including the core, the vessel upper plenum, the hot legs, the steam generator U-tubes, the cold legs with the loop seal and the pumps and the down-comer and the lower plenum of the vessel, can be characterized by:

- A heat source with power around 3% of the nominal value of the NPP.
- A heat sink (constituted by SG secondary side) having the same power removal capability as above.
- The pressure in the range 7-9 MPa.
- The coolant temperature close to saturation at about 300 °C.
- The (Natural Circulation) core flow-rate in the range 3-10 % of the nominal value.

All of this ensures a steady state and a possibly infinite stable working condition for the NPP (so called single phase natural circulation). Let us assume now that the primary coolant is removed in a step-like manner with steady conditions established after each draining period. This corresponds to the quasi-steady small break LOCA (Loss of Coolant Accident) conditions; in this situation, the following NC regimes can be distinguished in a hypothetic time sequence:

- ➔ Single-phase natural circulation (i.e. the condition depicted by the bullets above, before the draining start).
- ➔ Two-phase natural circulation – stable.
- ➔ Two-phase natural circulation – unstable or siphon-condensation.
- ➔ Reflux condensation.
- ➔ Occurrence of fuel rod dry-out (or steam superheating at core outlet).

The five NC regimes are characterized below, see also D'Auria & Frogheri 2002.

Single Phase Natural Circulation

Single Phase Natural Circulation (SPNC) regime implies no saturated void occurrence in the upper plenum of the system. Therefore, coolant at the core outlet shall be sub-cooled up to nearly saturated. Core flow-rate derives from the balance between driving and resistant forces. Driving forces are the result of fluid density differences occurring between [descending side of U-Tubes & vessel down-comer] and [core & ascending side of U-Tubes]. Resistant forces are due to irreversible friction pressure drops along the entire loop. Resulting fluid velocities are sufficient for removing core power in (sub-cooled) nucleate boiling or forced convection heat transfer regimes: no film boiling condition is experienced in the core. It may be noted that the secondary side of SG is also a natural circulation system working in two-phase conditions.

SPNC may occur at any primary system pressure, consistently with SG pressure. However, in transient NPP conditions primary system pressure ranges between 0.2 and 16 MPa and NC may occur at any pressure value. When secondary pressure is close to the nominal operating condition and no or very small leakage is present in primary circuit, the primary system pressure stabilizes around 8 MPa as already mentioned. It is expected from the NPP design that SPNC, provided the availability of SG cooling, is capable to remove the nuclear heat decay from the core. The available experimental database including NPP tests confirms this capability.

Stable Two-Phase Natural Circulation
Two Phase Natural Circulation (TPNC) regime occurs as a consequence of coolant loss (or draining) from the primary system. Both driving and resistant forces increase when decreasing mass inventory of primary system, owing to the presence of two phase flow including boiling and condensation phenomena. Occurrence of two-phase flow implies saturated void appearance (i.e. boiling) in upper plenum and in hot legs. Presence of voids in pressurizer and in upper head is considered to keep the system (core cooling) in single phase conditions.

Assigned the typical geometrical layout of PWR, the former effect, i.e. larger increase of driving forces, is prevailing at small decreases of coolant mass inventories. The latter effect, i.e. larger increase in resistant forces, is prevailing for large decreases of coolant mass inventories. The net result is a 'peak' in core mass flow-rate versus primary system inventory (when primary mass flow-rate decreases), as can be observed in the flow regimes maps reported in Fig.1 (see discussion below).

Forced convection, sub-cooled and saturated heat transfer regimes are experiences in the core. Condensation occurs inside the U-Tubes of steam generators (primary side). The average core void fraction is typically less than 30%, whereas at the outlet values ranging around 50% can be reached without occurrence of thermal crisis in the considered pressure range.

Siphon Condensation
When the primary mass inventory is decreased below certain limits (see below), the simultaneous occurrences of

- decreasing of NC driving forces,
- small temperature difference across U-Tubes of steam generators,
- condensation (mostly) in the rising part of the U-Tubes in primary side,
- Counter Current Flow Limiting Phenomenon (CCFL) at the entrance of U-Tubes, e.g. see D'Auria et al. 1994,

are at the origin of wide system oscillations of core inlet flow-rate. The phenomenon has been investigated by Kukita & Tasaka 1989, and Aksan et al. 1995, in the last case based on a natural circulation experiment performed in Lobi facility, (test A2-77a, D'Auria. & Galassi, 1990). Evidences of the phenomenon have been found also in other facilities.

At mass inventories of the primary system around 70% of the nominal value, the efficiency of the condensation heat transfer across U-Tubes causes the release of almost all core thermal power in the ascending side of U-Tubes. Liquid level builds up and is prevented to drain down by the steam-liquid mixture velocity at the tube entrance, i.e. the CCFL condition occurs. Therefore, liquid level rises in the U-Tubes till reaching the top. During this period, typical duration of the order of 10 s, flow-rate at core inlet is close to zero and core boil-off occurs, without achieving the dry-out condition. Once the liquid level reaches the upper bend of U-Tubes, the siphon effect occurs and causes the emptying of the ascending side of U-Tubes and the re-establishment of core inlet flow-rate. A new cycle starts. The phenomenon is made more complex by the interaction of flows inside the thousands of U-Tubes that constitute a SG tube bundle. Different groups of tubes may stay at a different stage of the oscillation at the same time, also causing flow reversal in the tube bundle. Suitable core cooling still can be achieved in these conditions. The phenomenon has been characterized as ‘siphon condensation’ by D’Auria & Galassi 1990 and 1990a. Additional information can be found in D’Auria & Galassi 2007, as given in the present Course.

Reflux Condensation

At ‘low’ mass inventories of primary coolant and/or at low core power, steam velocities in the upper part of the system including hot legs and steam generator entrance are low. Weak interactions occur at the steam-liquid interface that is not enough to cause CCFL. In these conditions, the liquid that is condensed or entrained in the ascending side of the U-Tubes may flow back to the hot leg and to the core.

Thus, stratified countercurrent steam and liquid flow occur simultaneously in the hot legs. Mass flow-rate at core inlet is close to zero, although a ‘minor’ natural circulation path may establish between core and down-comer inside the vessel. However, upward two-phase mixture and downward liquid flows occur at the core outlet. Core thermal power can be removed by boil-off in the saturated nucleate boiling heat transfer regime.

Dry-out occurrence

Dry-out is a singular point (or region) in the two-phase heat transfer curve: it is related to core region and implies a steam blanket in contact with the heat source. Therefore dry-out shall not be considered a natural circulation regime, however, the terms ‘dry-out occurrence’ appear in the right part of Fig. 1 (see below) and, above all, in the NC scenario depicted here dry-out occurs when primary system mass inventory is roughly lower than 40% of the nominal value.

Dry-out is caused by the combination of low flow and high void fraction. As a consequence, film boiling heat transfer regime is experienced on the fuel rod claddings with low coefficient for convection heat transfer. Rod surface temperature increases in various zones of the core and the overall process of thermal power transfer from fuel rods to the fluid may become unstable. The system operation in these conditions is not acceptable from a technological point of view.

It may be noted that the temperature excursion is strongly affected by primary system pressure and thermal power levels: the linear rod power plays a role in these conditions. At primary system pressure around 15 MPa (nominal operation for PWR) ‘post dry-out’ surface temperature jumps may be as low as a few tens of Kelvin, tolerable for the mechanical resistance of the rod-clad material. Vice-versa, temperature jumps may results as high as several hundreds Kelvin at low pressure.

The flow regime map

Based on the information provided above, the natural circulation flow regime map in Fig. 1 has been built. In the vertical axis the flow-rate at core inlet is reported in the units of % of the initial nominal value corresponding to 100% core (or system in the case of ITF) power. In the horizontal axis the coolant mass inventory in primary system is again reported in % of the initial nominal value. The diagram is obtained assuming decay power and a pressure in primary system close to the saturation pressure corresponding to coolant temperature at nominal conditions (actually and average between hot and cold leg temperatures). Therefore, the point on the vertical axis (i.e. at 100 mass inventory) corresponds to the NC flow-rate value in a PWR system when the main coolant pumps have been tripped off, the core has been scrammed and the secondary side of steam generators is kept at nominal (or close to) pressure with feed-water available at the same rate (in %) as the core power.

The full line with open dots has been derived as a hand made qualitative approximation considering data measured in ITF and results of computer code calculations, as discussed in D’Auria et al. 1991. The specific measured data and the results from predictions are indicated by different dots in the legend on the right of the figure and additional information can be found in the listed references.

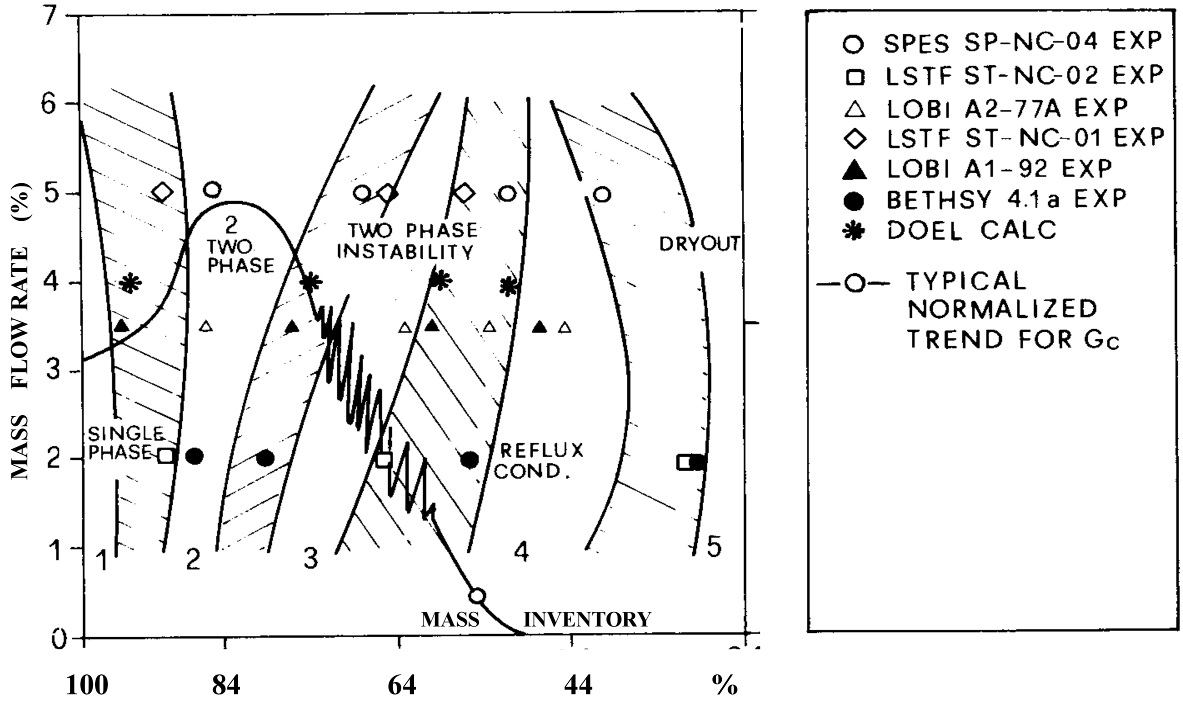


FIG. 1 – The NC flow regime map: characterization of the natural circulation phenomena in PWR systems.

The curve in the figure allows the identification & characterization of the five 'arrow' NC regimes discussed at the beginning of the chapter, i.e. in sequence (with decreasing primary system mass inventory): single phase NC, two-phase stable, two-phase unstable, reflux condensation and dry-out occurrence.

The single phase NC ends with the emptying of the pressurizer. Soon after, following the scale in the abscissa, NC flow-rate increases to a maximum when primary system mass inventory achieves 85% value. In the decreasing region of the NC flow-rate curve, siphon condensation instabilities occur causing lower peaks also in the negative range for mass flow-rate (not shown in the figure). Reflux condensation is characterized by nearly zero net flow at core inlet (bottom, i.e. connection between lower plenum and core); however, upward and downward flows of two-phase mixture and of (nearly) saturated liquid occur at the core outlet (top, i.e. at the connection of core and upper plenum). Finally, depending on the various considered systems, dry-out occurs when the residual mass inventory ranges between 25% and 40% of the nominal value.

All flow regimes are roughly characterized, as demonstrated by the wide transition regions shown in Fig. 1 (shadowed areas).

A few diagrams supporting the description given for the NC scenarios in PWR and for the 'synthesis' NC regimes map (Fig. 1) can be found in Figs. 2a to 2d, D'Auria et al. 1991. Fluid velocity measurements taken in hot and cold legs (Fig. 2a, with decreasing coolant mass inventory as a function of time) show the qualitative trend of core inlet flow-rate including the increase in two phase region and the oscillations (siphon-condensation). A detail taken from pressure drop measured signal during the oscillatory period (siphon condensation) can be seen in Fig. 2b. The normalized 'G vs RM' (core inlet flow-rate versus Residual Mass inventory) can be seen in Fig. 2c as measured in three different scaled ITF: a scale effect on can be observed (K_v is the volume scaling ratio increasing along the arrow inn Fig. 2c). The mass inventory at which dry-out occurs (Fig. 2d) is apparently not affected by the size of the PWR system (NPP either ITF).

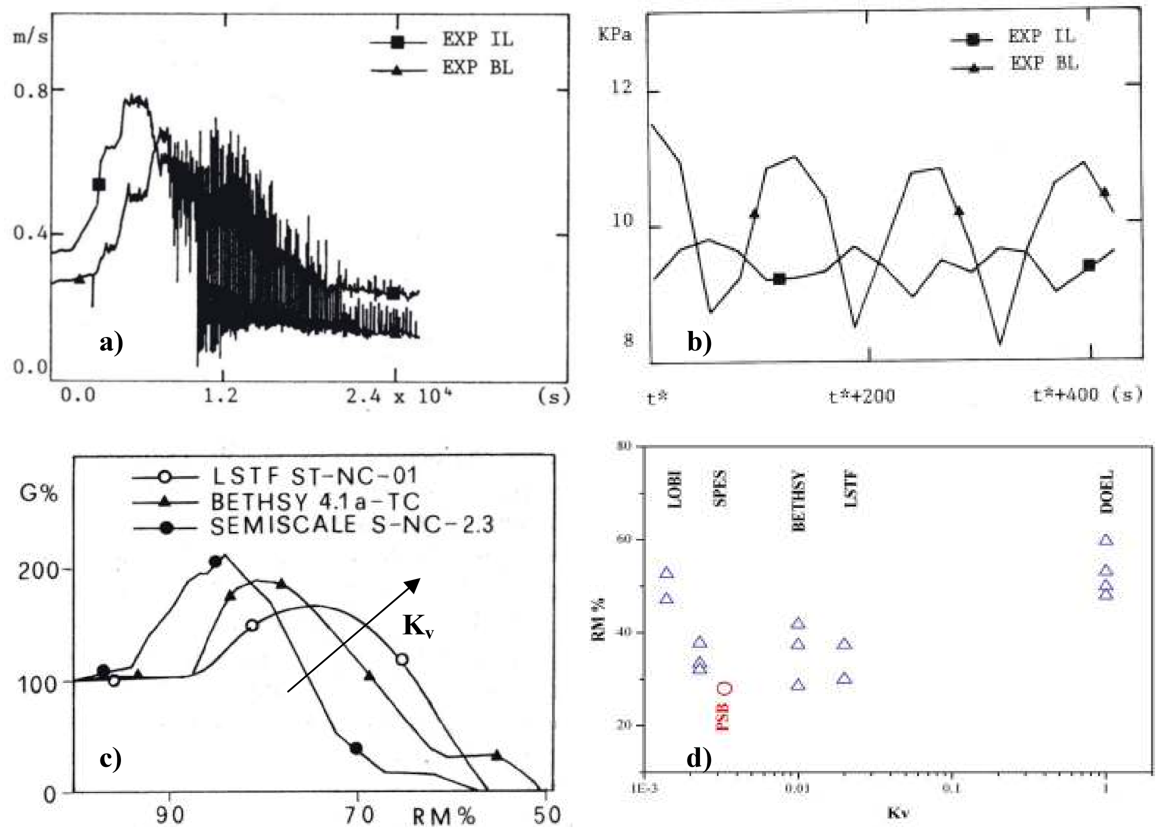


FIG. 2 – Natural circulation phenomenology in PWR: a) fluid velocity in hot leg and cold leg; b) differential pressure at inlet of steam generators; c) effect of volume scaling upon the NC curve; d) dry-out occurrence as a function of mass inventory in different PWR systems (ITF and NPP).In relation to PSB, see additional remarks in section 4.2.

3. THE NATURAL CIRCULATION FLOW MAP

The similitude in the geometry and in the operational characteristics of the PWR simulators, or Integral Test Facilities (ITF), mentioned in the chapter above allowed a direct comparison between results of NC experiments. The database gathered from ten experiments performed in the six ITF listed in Tab. 1 has been used by D'Auria et al. 1994a and D'Auria et al. 1998 to establish a Natural Circulation Flow Map.

TAB. 1 RELEVANT HARDWARE CHARACTERISTICS OF THE PWR SIMULATORS CONSIDERED FOR NATURAL CIRCULATION.

Item	**1 Semiscale Mod2A**	**2 Lobi Mod2**	**3 Spes**	**4 PKL-III**	**5 Bethsy**	**6 Lstf**
Reference reactor and power (MW)	W-PWR 3411	KWU-PWR 3900	W-PWR 2775	KWU-PWR 3900	FRA-PWR 2775	W-PWR 3423
No of fuel rods simulators	25	64	97	340	428	1064
No of U-tubes per SG	2/6	8/24	13/13/13	30/30/60	34/34/34	141/141
Internal diameter of U-tubes (mm)	19.7	19.6	15.4	10.0	19.7	19.6
Actual Kv	1/1957	1/589	1/611	1/159	1/132	1/48

The ITF in Tab. 1 have been designed according to the scaling criteria:

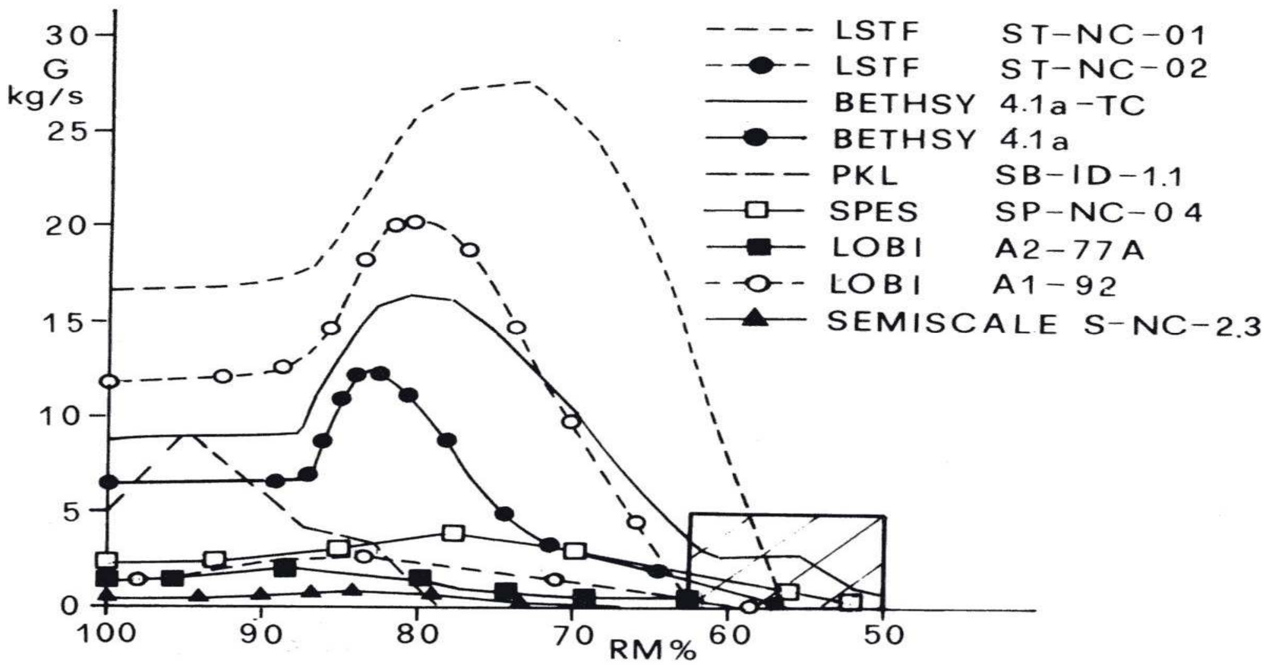
a) preserving (related to the reference NPP) the volume/power ratio,
b) preserving the height, the pressure and the linear power of fuel rod simulator,
c) preserving the equivalent diameters and the pressure drops as much as possible,
d) preserving the time of occurrence of main events.

The above conditions imply the same coolant temperature (i.e. as in the reference NPP), the same coolant velocities and the flow-rates scaled like the volumes. A number of 'scaling distortions' are unavoidable: e.g. quantities like transit time in the loop for a volume-of-fluid, response time of fuel simulators, thermal power release from structures and flow areas cannot be scaled down according to scaling theory requirements. Furthermore, in the last row of Tab. 1, the volume ratio for each ITF is 'normalized' related to the largest ITF (Lstf in the table).

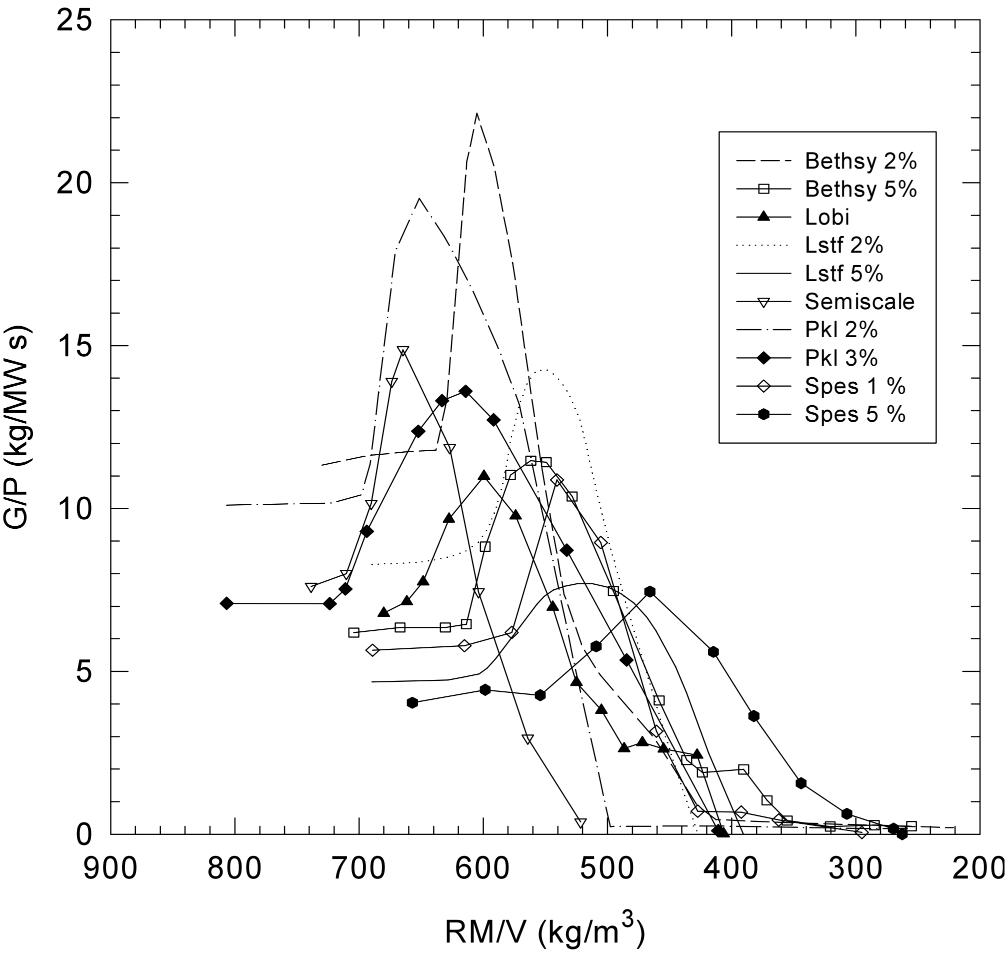
In all the considered ITF, NC experiments with similar modalities have been performed and the NC regimes discussed in the previous chapter are experienced. The linear power of fuel rod simulators, the fraction of nominal core power and the primary system pressure constitute the main differences for the boundary conditions of the considered experiments. In relation to pressure, PKL experiments have been performed at a pressure value roughly one half the values adopted in the other facilities. The range of design and operational parameters of the ITF (e.g. pipe diameter, system volume, number of steam generators, heat losses to the environment), not explicitly discussed here, and the identified differences are assumed to produce an envelope for any expected NC situation in a typical PWR when decay heat removal is concerned.

Measured values of core inlet flow-rate (G, Kg/s), core power (P, MW), primary system fluid mass inventory (i.e. Residual Mass, RM, Kg) and net volume of the primary system (V = const., m^3), have been used for setting up the Natural Circulation Flow Map (NCFM). The diagram G/P versus RM/V has been preferred for the NCFM over other possible choices including non-dimensional quantities.

The experimental database from ITF (namely, six facilities and ten experiments) and the envelope of curves are given in Figs. 3 and 4, respectively. The envelope of curves in Fig. 3b (reported in Fig. 4 for the clarity of the presentation) is assumed to constitute the NCFM of PWR at decay core power and pressure corresponding to the saturation values at coolant temperature ranging around nominal operational values in cold and hot legs of PWR. It shall be noted how the physical unit change from kg/s versus % system mass, Fig. 3a, to G/P (kg/s Mw) versus RM/V (kg/m^3), Fig. 3b, changes the mutual comparison among NC curves measured in different ITF.



a)



b)

FIG. 3 – Natural Circulation system behaviour measured in ten experiments performed in six PWR simulators (ITF): a) core inlet flow-rates versus mass inventory in primary loop; b) G/P versus RM/V..

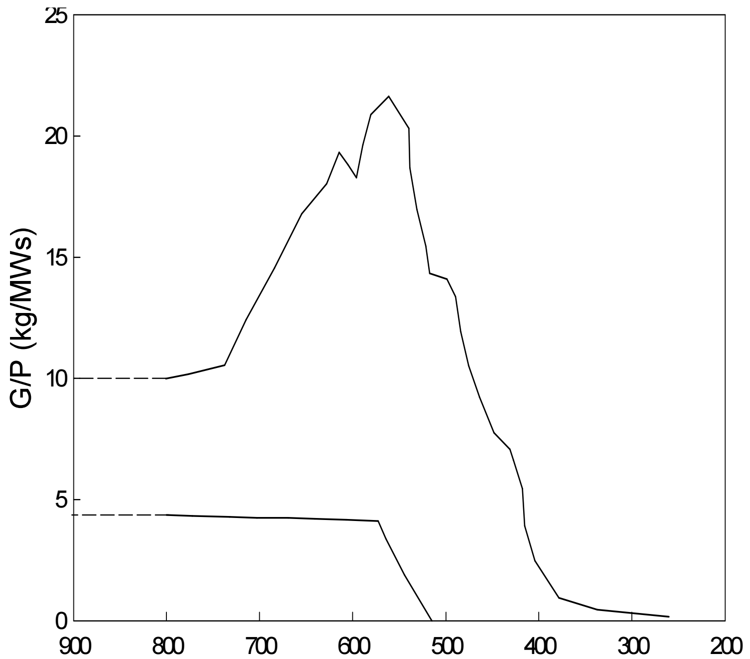


FIG. 4. – Natural Circulation Flow Map derived from the envelope of measured NC flow-rates curves in PWR simulator.

4. THE APPLICATION OF THE NCFM

The NCFM is expected to constitute a boundary for the relevant NC parameters in PWR type systems including NPP. When a (PWR) system exhibits a natural circulation curve in the plane G/P vs RM/V above the upper boundary it can be concluded that the system is well designed for NC. The opposite is true when the G/P vs RM/V curve of the considered system lies in the bottom part (or outside of the bottom) of the NCFM. Thus the NFCM can be used to evaluate the NC Performance (NCP) of (PWR) systems. This is specifically helpful for the design of innovative reactors. It is expected that innovative reactors (e.g. IRIS, but also AP-1000 and EPR) have a better NCP than existing PWR.

Two applications of the NCFM are distinguished hereafter to evaluate the NCP of PWR systems: the former embedded into the paper by D'Auria & Frogheri 2002, the latter dealing with latest studies.

4.1 NCFM applications by D'Auria & Frogheri 2002

Seven commercial NPP systems and three ITF, not used for setting up the database presented in Fig. 2, were adopted for demonstrating the use of the NCFM. Main characteristics of the NPP and of the ITF can be drawn from Tabs. 2 and 3, respectively.

Reactors 1 to 4 (Tab. 2) have been built and are in operation. Reactors 5 to 7 are in a more or less advanced design stage. The geometric layout of primary systems for reactors 1, 2, 5, 6 and 7 is similar to the geometric layout of ITF originating the database for the NCFM. However, differences are present in the relative elevations between core and SG. Reactors 3 and 4 are equipped with OTSG and HTSG, respectively. So the geometric layout of the primary system is different from the geometric layout of ITF originating the database for NCFM.

Pactel and RD14m (Tab. 3) are experimental simulators of WWER-440 and CANDU NPP, respectively. Their geometric layout is different from those of a PWR. In the case of WWER-440, six loops equipped with horizontal tubes steam generators are connected to the vessel, though only three are simulated in Pactel facility. The horizontal core configuration characterizes the CANDU design, that otherwise is equipped with U-Tubes steam generators: both characteristics are present in the RD-14m facility.

TABLE 2. RELEVANT CHARACTERISTICS OF NPP CONSIDERED FOR THE APPLICATION OF THE NCFM.

	1 PWR	**2 PWR**	**3 PWR**	**4 WWER-1000**	**5 EPR**	**6 AP-600**	**7 EP-1000**
Nominal Power (MW)	1877	870	2733	3000	4250	1972	2958
Primary System volume (m3)	167	150	330	359	459	211	339
SG type	U-Tubes	U-Tubes	Once-Through	Horizontal	U-Tubes	U-Tubes	U-Tubes
No. of loops	2	4	2	4	4	2	3
No. of pumps	2	4	4	4	4	4	6
Nominal mass inventory (Mg)	108	108	224	240	307	145	227
Nominal Core Flow (Kg/s)	9037	3150	17138	15281	20713	8264	14507
Pressurizer and SG pressure (MPa)	15.6 6.	14.0 3.1	15.0 6.4	15.7 6.3	15.5 7.2	15.5 5.5	15.8 6.4

A comparison has been made between measured (case of ITF) and calculated (case of NPP) system behavior during NC and the data that characterize the NCFM. Results of code calculations assuming stepwise draining of primary system fluid mass inventory have been performed (case of NPP) and relevant NC experimental data are considered (case of ITF).

TABLE 3. RELEVANT CHARACTERISTICS OF PACTEL and RD14M CONSIDERED FOR THE APPLICATION OF THE NCFM.

ITEM	Pactel (original)	Pactel (with CMT) °	RD14M
Reference reactor and power (MW)	WWER-440 1375	WWER-440 1375	CANDU 1800
No. of rods	144	144	70
No. of SG	3	3	2
SG type	Horizontal	Horizontal	U-Tubes
Actual Kv +	1/433	1/462	1/378

° CMT = Core Make-up Tank
+ Definition introduced for database in Table 1.

In the case of NPP, the qualification level of the adopted code and nodalization affects the calculated NCP. This shall be accounted for in the evaluation of the results. Furthermore, in the case new generation 'passive' safety reactors 6 and 7, the emergency loops connected with the primary system are assumed to come into operation once the coolant draining process is initiated.

Calculated or measured transient scenarios reflect the NC flow regimes identified in Fig. 1. The reflux condensation NC regime could not be observed in the systems equipped with horizontal and "once-through" steam generators and is not evident from the RD14M experimental database. The siphon condensation NC regime is also not evident in all the calculations or available from the considered experimental databases. However, siphon condensation is calculated for the NPP 6 and 7 in Tab. 2.

Significant results are shown in Figs. 5 to 7. The following observations apply:

- Natural Circulation Performance (NCP) of UTSG equipped PWR and of WWER-1000 is qualitatively similar (Fig. 5), see also Mousavian et al. 2004. The good performance of the PWR-2 can be noted (low power NPP equipped with four UTSG).
- The OTSG equipped PWR show an 'early' NC flow-rate decrease and an early stop of NC due to void formation in the 'candy-cane' and the rising part of the hot leg (Fig. 5).
- NC flow-rate in AP-600, as expected, is not affected by draining because of liquid mass supplied by the 'passive' emergency cooling loops (Fig. 6). This behavior does not show up in the case of EP-1000 (lack of qualification of the adopted code model).
- The WWER-440 simulator (Pactel, see also Tuunanen et al., 2000) exhibits a decrease of the NC flow-rate at relatively high mass inventories of the primary loop. The presence of the hot leg loop seal is at the origin of partial flow stagnation (Fig. 7). Removal of the hot leg loop seal is effective in improving NCP as shown by the calculated WWER-1000 transient. The consideration of a passive system also improves the NCP of the Pactel.
- The CANDU simulator RD-14m (e.g. see Richards et al., 2003) exhibits two behaviors depending upon the considered experiment (Fig. 7). This shows the need of a deeper investigation before drawing conclusions. It may be noted that larger driving forces characterize CANDU systems for NC compared with PWR systems, owing to the larger distance of heat source and sink. However, larger pressure drops are also expected owing to the longer core and to the presence of small equivalent diameter pipes at core inlet and outlet.

Any attempt to judge the achieved results, i.e. the NCP of involved NPP and ITF, should consider the quality of the used data bases (DB). The result of a two step evaluation can be found in Tab. 4. The second column considers the demonstration, if available, of the quality of the starting DB. If 'N' appears in the second column, the evaluation in the third column is not meaningful. The third column considers the quality demonstration of the used DB, e.g. the nodalization in the cases of code use and, finally, the NCP judgement. Again, if 'N' appears in the quality demonstration, the last evaluation is not meaningful.

The NS for PWR-3 is due to the poor NCP also at high values of the RM inventory caused by the long vertical hot leg. Quality of data is assumed to be suitable. The 'N' in the case of EP-1000 (third column) is related to the available nodalization. The dash (-) in the case of RD14m comes from the contradictory available information in relation to RD-14m (see section 4.2).

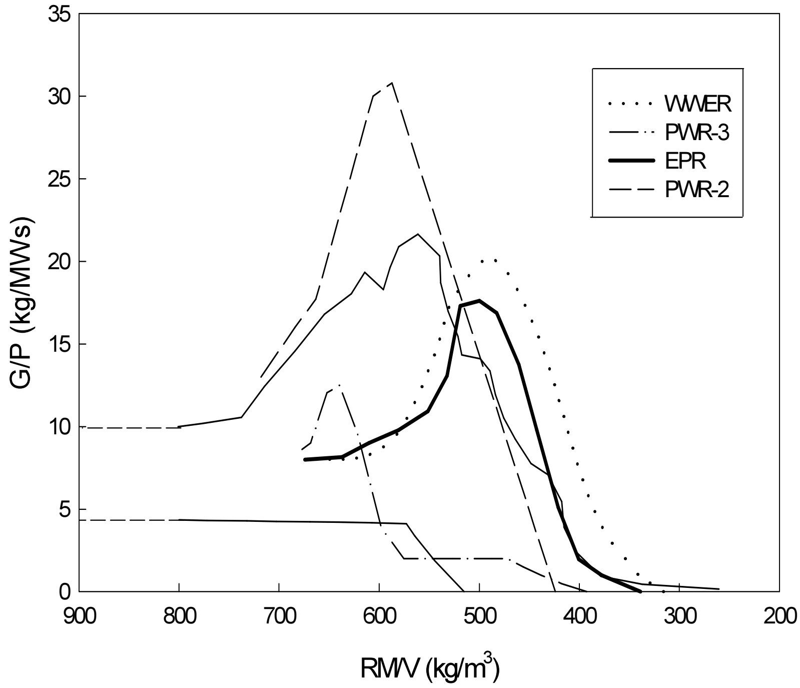


FIG.5 – Evaluation of the NCP for PWR-1, PWR-2, WWER-1000 and EPR NPP (Tab. 2) by using the NCFM.

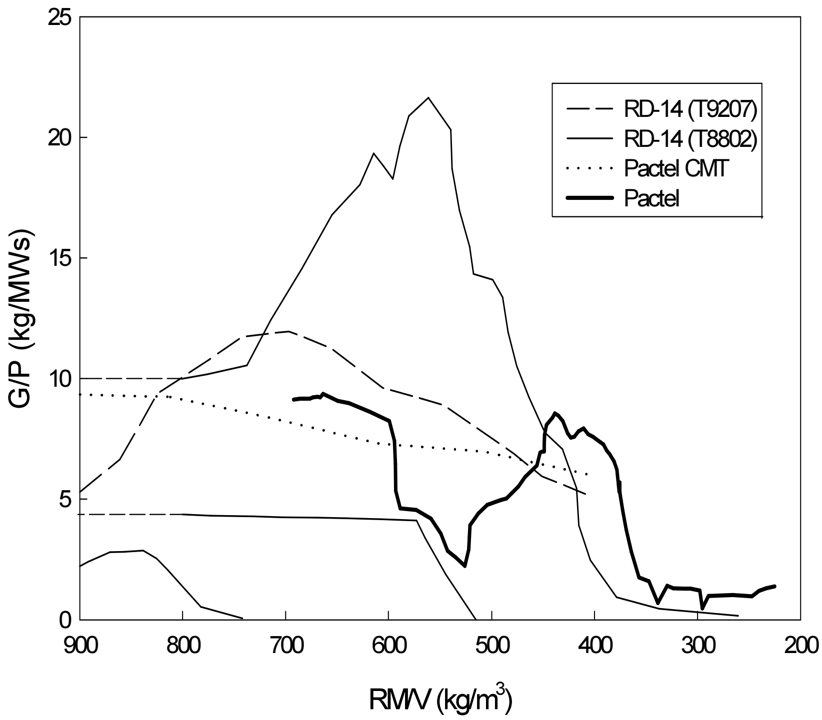


Fig. 6 – Evaluation of the NCP for PWR-3, AP-600 and EP-1000 NPP (Tab. 2) by using the NCFM.

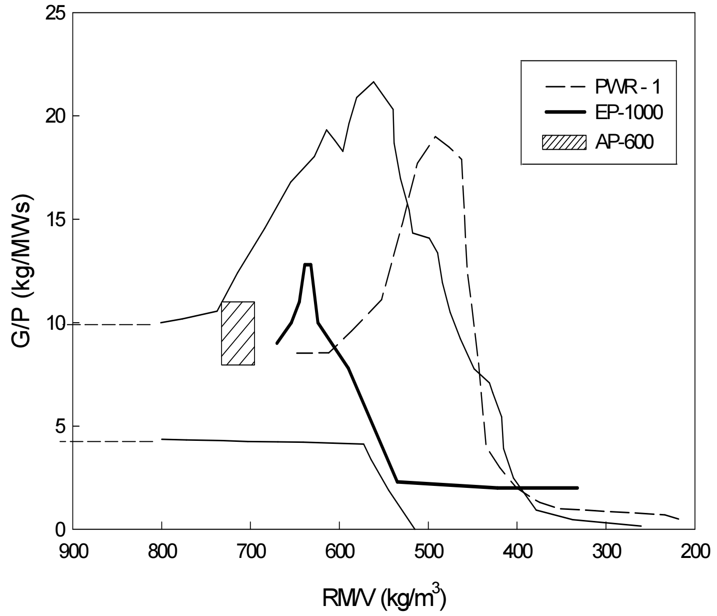


Fig. 7 – Evaluation of the NCP for Pactel and RD-14m (preliminary) NPP simulators (Tab. 3) by using the NCFM.

4.2 Additional applications of the NCFM

Four main additional applications of the NCFM have been completed so far and are outlined hereafter:

a) Demonstration that the ATUCHA-1 PHWR NPP has a good Natural Circulation Performance, e.g. Ferreri et al. 2003.
b) Evaluation of the NC experiment performed in PSB-VVER installed in EREC research center near to Moscow, e.g. Cherubini et al., 2005
c) Evaluation of the NC experiment performed in the PKL-III facility (PWR simulator) installed in Erlangen, e.g. Araneo et al., 2005.
d) Calculation of NC for the CANDU simulator RD-14m obtained by a 'qualified' code-nodalization, e.g. D'Auria and Galassi, 2002, and D'Auria et al., 2006, see also Richards et al.2003 and Prosek et al., 2004.

TAB. 4 – SUMMARY OF NCP EVALUATION FOR THE CONSIDERED NPP AND ITF.

NPP or ITF	QUALITY OF STARTING DB	QUALITY OF DB &NCP EVALUATION
PWR-1	Y	Y & S
PWR-2	N	Y & S
PWR-3	Y	Y & NS
WWER-1000	Y	Y & S
EPR	N	Y & S
AP-600	Y	Y & S
EP-1000	N	N & -
Pactel	Y	- & S
Pactel with CMT	Y	- & S
RD14M	N.A.	-

Y Confirmed **N Not Confirmed**
N.A. Not Available **- Not applicable**
S Suitable (for NC) **NS Not Suitable**

NC calculation in Atucha-1 NPP

The NCFM was used to characterize the behavior of the CNA-I PHWR NPP in NC flow conditions, in a reduced primary mass inventory scenario. One reason for the analysis was to demonstrate the similarity of Atucha-1, channel type pressurized heavy water reactor (PHWR), and a typical PWR in relation to NC. The core active length of Atucha-1, i.e. more than 5 m, constitutes a further motivation for the study.

The simulations have been performed using three different nodalizations with increasing level of detail:

- The nodalization identified as SET-I (both nodalization and calculation are identified by the same label) is the coarser one and represents a plant layout similar to a PWR. Namely the moderator system was hydraulically disconnected from the primary circuit of Atucha-1. This permitted to verify that the trends known for most ITF working in similar situations give an envelope to the CNA-I behavior, considering appropriate activation of selected NPP systems.
- The calculation SET-II was performed by considering the hydraulic connection of the moderator system and the primary system.
- The calculation SET-III was performed by a fine nodalization developed by the utility owner of the Atucha-1 NPP. Furthermore, in order to improve the consistency with the PWR experimental database, the thermal power exchange between primary system and moderator system was 'fictitiously' imposed (i.e. not representative of the real situation) to be proportional to the core generated power (and not obtained by solving the energy balance equations including the primary circuit and the moderator system).

Selected relevant nominal operating parameter values for the CNA-1 are provided in Tab. 5. The results from the study are given in Figs. 8a to 8c obtained from calculation/nodalization SET-I, SET-II and SET-III, respectively. Different calculation were performed for each nodalization as detailed by Ferreri et al., 2003. In general terms, the results proved the adequacy of the NC in the Atucha-1 NPP.

TABLE 5. NOMINAL PARAMETER VALUES OF THE ATUCHA-1 NUCLEAR POWER PLANT.

CNA-I Overall Plant Data	
Reactor type	PHWR
Net power station output	~ 345 MWe
Reactor Coolant System and Moderator System	
Total thermal power	1179 MW
Number of fuel channels	253
Active core length	5300 mm
Shape of fuel assembly	37-rod cluster
Reactor Coolant System and Moderator System	
Coolant and moderator	D_2O
Total thermal power transferred to the feed water/main steam cycle	1186 MW
Total thermal power transferred to steam generators	1076 MW
Total thermal power transferred to moderator coolers	110 MW
Number of coolant circuits	2
Number of moderator circuits	2
Total coolant circulation flow	6000 kg/s
Total moderator circulation flow	440 kg/s
Pressure at reactor vessel outlet	114 bar
Coolant temperature at reactor pressure vessel	573 K
Average moderator temperature normal/maximum	443 K/ 493 K
Steam pressure at steam generator outlet	4.4 MPa
Total steam flow	510 kg/s

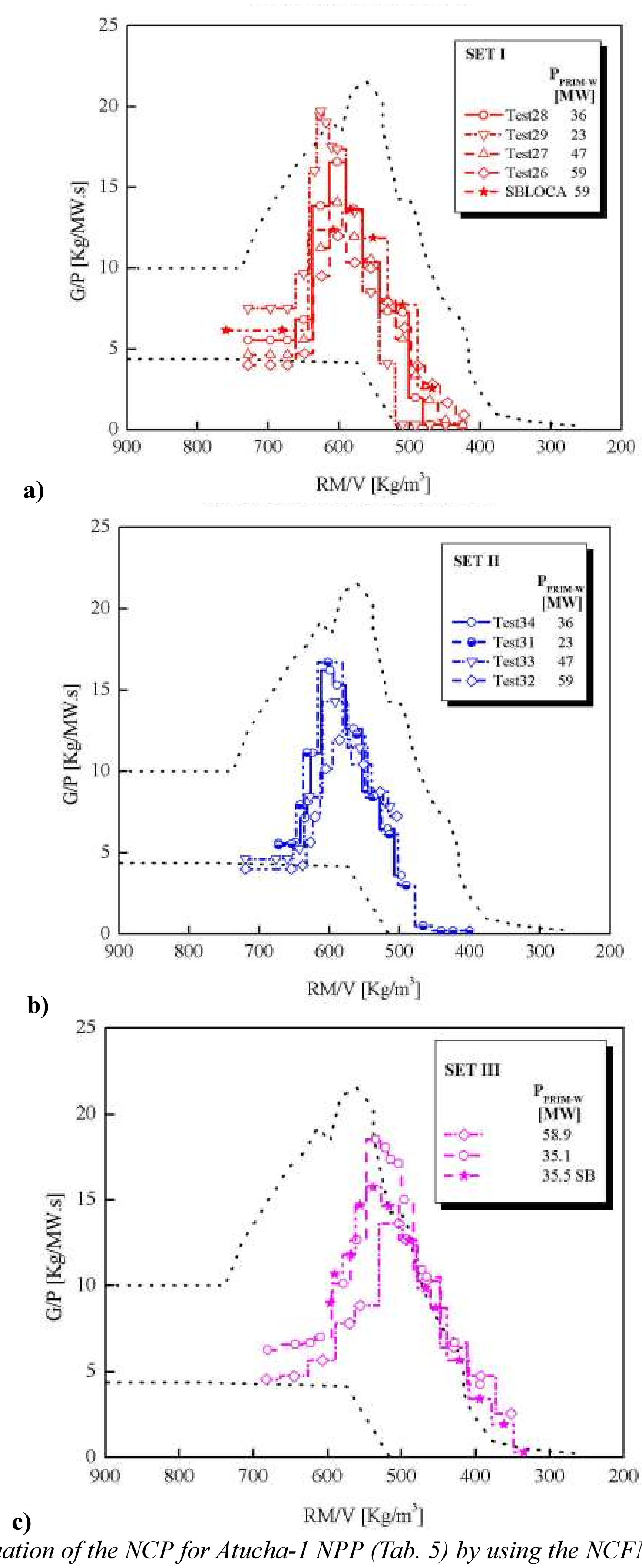


FIG. 8. Evaluation of the NCP for Atucha-1 NPP (Tab. 5) by using the NCFM: results from three different nodalizations, a) SET-I, b) SET- II and c) SET- III.

NC experiment and calculation in PSB-VVER

The PSB-VVER simulates in the volume and power scale of 1/300 the VVER-1000 NPP and is installed in the Electrogorsk EREC center near Moscow. Height and pressure (and then coolant temperature) scaling are equal to unity. Thus it is expected the real time simulation for accident scenario evolutions in the prototype NPP. A NC experiment was performed and identified as NC-1 having the same characteristics of the NC experiments utilized in the NC database, namely test duration was about 21000 s when dry-out occurred at a quite low value of the primary system mass inventory as shown in the diagram of Fig. 2d.

The outcomes of the study are given in Fig. 9 where NC experimental data and results of Relap5 calculations are shown. Comparison between experiment and prediction show an overestimation of the calculated data in the rising zone of the NCFM upper boundary. This is due to 'minor' inadequacies in setting the values of local loss coefficients in the PSB-VVER nodalization. Furthermore, at low values of the primary system mass inventory the experimental data show a deviation from both the calculation and the upper bound line of the NCFM. This is due to the steam flowing across the fluid velocity transducer that produces higher flow-rate values than actual ones. The good NCP for the VVER-1000 system (experimental data in ITF in the present case) can be deduced thus confirming the findings in section 4.1, see also the paper by Mousavian et al. 2004.

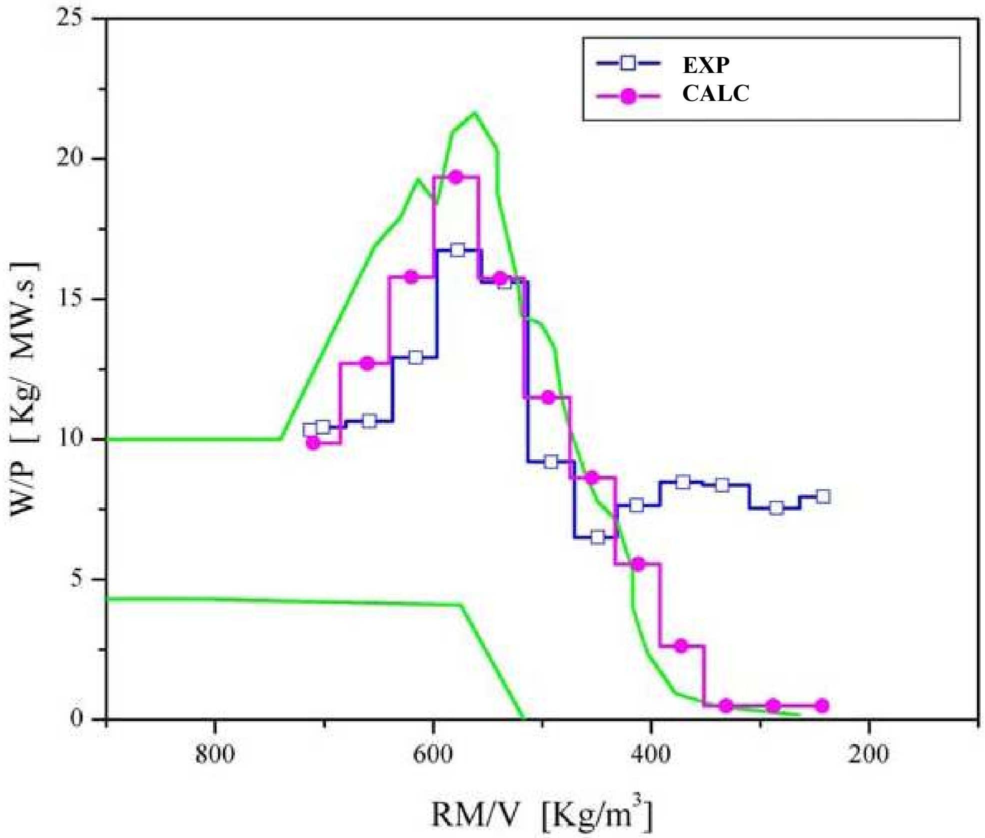


FIG. 9. Evaluation of the NCP for PSB-VVER ITF by using the NCFM: experimental data and results of calculations are reported.

NC experiment in PKL-III

PKL-III is a PWR simulator ITF installed at Erlangen in Germany, with prototype being the Biblis nuclear reactor. This is an upgraded version of the PKL facility discussed in Tab. 1, see also Del Nevo et al. 2006. The facility has been recently used for boron dilution experiments, see also D'Auria & Galassi 2007. A natural circulation experiment was performed within the OECD framework and identified as test F1.2 (restricted distribution). The test was performed with primary circuit at a pressure around 4 Mpa and saturated conditions in secondary side of steam generators. The test modalities, involving draining of primary coolant were the same as used for deriving the NCFM.

The use of the NCFM, Fig. 10, (only experimental data shown, with results of Cathare analysis available by Del Nevo et al. 2006) shows the 'expected' suitable NCP of the facility.

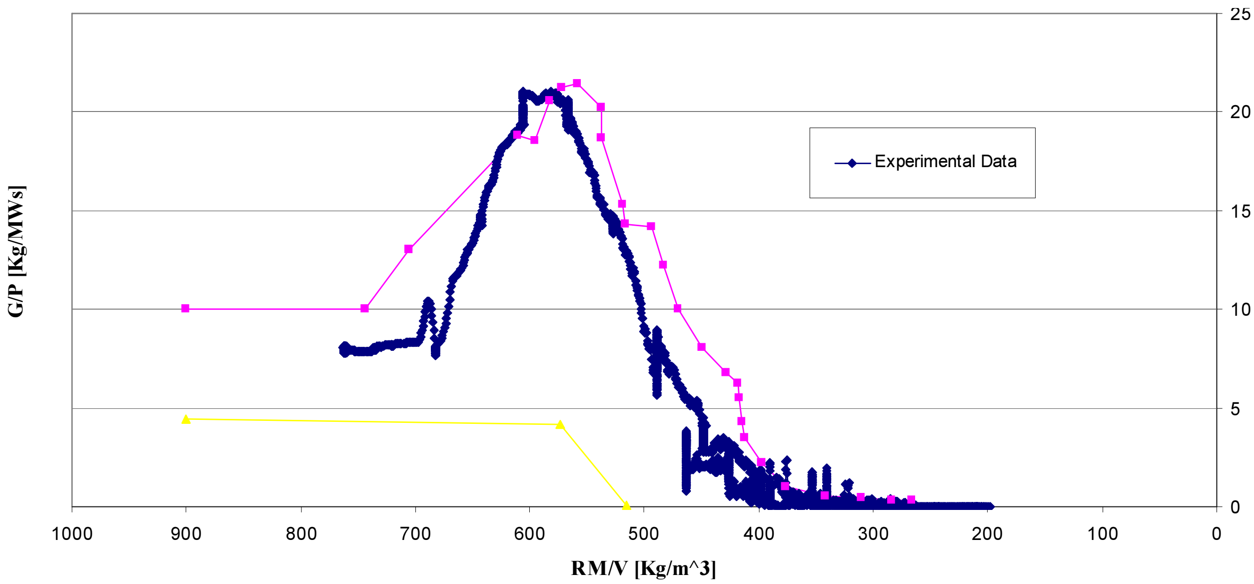


FIG. 10. Evaluation of the NCP for PKL-III ITF by using the NCFM: experimental data are reported.

<u>NC calculation in RD-14m</u>
The evaluation of the NC performance of CANDU type systems has been an issue since the development of the NCFM as also shown in Fig. 6, see also D'Auria et al.1995. The comparison between a PWR and a CANDU system as far NC is concerned should be based on the following aspects:

- The driving force for NC is largely affected by the distance in terms of elevation change between 'thermal centers' in the core region (i.e. the heat source) and in the steam generator (i.e. the heat sink) region. The above distance is comparable in the two different systems (PWR and CANDU) and even larger in CANDU.
- The resistant force, assuming similarity of steam generators (U-Tubes steam generators are installed in both PWR and CANDU systems), is largely affected by the length of the 'low equivalent diameter' of the core and of the connected regions. This length is much larger in CANDU and includes longer core region (with fuel rods) and exist and inlet piping connected with each fuel channel (till the respective headers).

Therefore, in the case of CANDU and related to a typical PWR considering a similar driving force and a much higher resistant force, in similar thermal-hydraulic conditions (e.g. pressures of primary and secondary systems and power per unit volume of fluid), one can easily expect a 'worse' NCP. This is what was derived from preliminary analyses of data in RD-14m (Fig. 6). However, in the case of CANDU the NC between core and steam generators can be combined, at least for an assigned time period, with channel to channel NC, being the entire group of channels thermally connected with the heat sink constituted by the moderator tank.

The availability of a qualified Relap5 nodalization of the RD-14m ITF (CANDU simulator), utilized for the analysis of the large break LOCA (Loss of Coolant Experiment) test B-9401 (see references listed above), suggested to perform a calculation of a NC scenario having the same characteristics (e.g. pressure, core power, draining modalities, steam generator performance) as the scenario adopted in PWR at the basis of the NCFM. The key results are given in Figs. 11 a) and 11b).

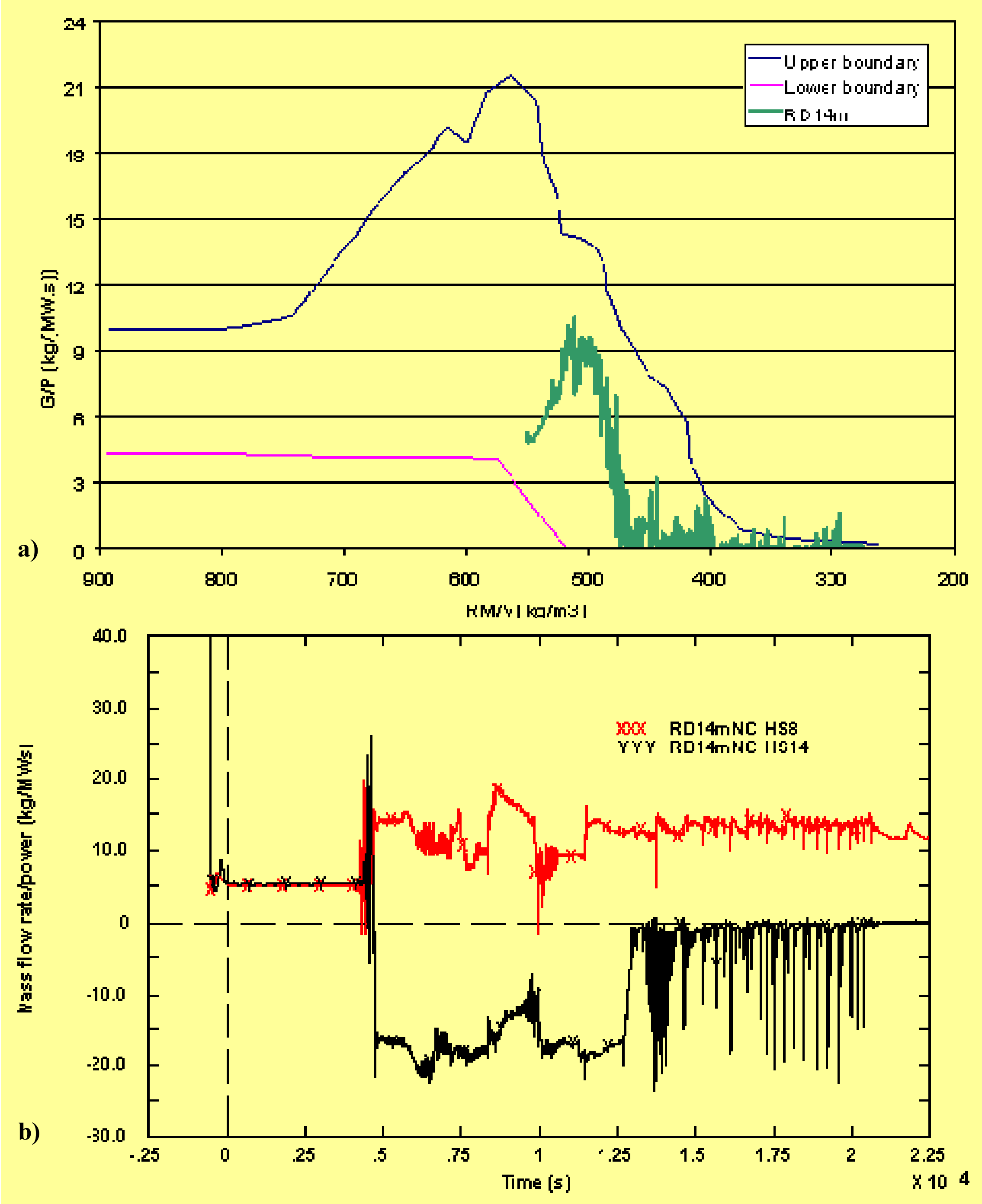


FIG. 11. Evaluation of the NCP for RD-14m ITF by using the NCFM, calculated data are reported: a) NCFM with calculated NCP; b) time trend, during the calculated NC scenario of mass flow-rates across two of the horizontal channels of the ITF .

The NCP in the considered CANDU system (Fig. 11a) is worse (though much better than the curve derived in Fig. 6 from the analysis of experimental data available in literature) than in a typical PWR, as far as the circulation between core and steam generators is concerned. Furthermore, the bifurcation point for the flow-rate in two selected fuel channels at about 5000 seconds in Fig. 11b), corresponds to the point where NC flow-rate between core and steam generators reaches the zero value in Fig. 11a) (i.e. at RM/V equal about 530 kg/m^3). This confirms the presence of the channel-to-channel NC that does not appear or does not constitute the subject in the NCFM.

5. CONCLUSIONS

The natural circulation (NC) scenario in PWR has been characterized following the analysis of measured and predicted data. This allowed the distinction between single-phase NC, two-phase NC, siphon condensation and reflux condensing and brought to the set up of a natural circulation flow regime map.

An experimental data base derived from six integral test facilities simulating PWR has been utilized to set up a reference natural circulation flow map. The map allowed the evaluation of performance in natural circulation conditions of pressurized water reactors or of system having the configuration of PWR, not necessarily of the same type as those used for deriving the map.

It was found that the PWR equipped with Once-Through steam generators and CANDU type systems have a poor natural circulation performance when primary mass inventory is decreased. In the case of CANDU, channel to channel NC is expected to occur and is not considered in the Natural Circulation Flow Map (NCFM). Otherwise, reasonable natural circulation performances of Russian designed reactors WWER were characterized. This is mainly true for the WWER-1000. Passive systems in the AP-600 innovative reactor are effective in keeping the primary system under single-phase natural circulation notwithstanding removal of coolant mass.

The NCFM reveals as a interesting instrument to judge the natural circulation performance of existing systems and of systems included in the forthcoming reactor generation.

REFERENCES

Aksan N., D'Auria F., Glaeser H., Pochard R., Richards C., Sjoberg A., 1995, "Separate Effects Test Matrix for Thermal-Hydraulic Codes Validation: Phenomena Characterization and selection of Facilities and Tests", *OECD/CSNI Report OCDE/GD (94) 82, Paris (F), Jan*uary.

Araneo D., D'Auria F., Del Nevo A., Galassi G.M., Giannotti W., Spadoni A., 2005, "Key Issues from the Cathare Assessment Against PKL-III Boron Relevant Transients", *Meeting DIMNP-EdF, Pisa July*.

Cherubini M., D'Auria F., Ferreri J.C., Mazzantini O., 2005, "Simulation of a Natural Circulation Test in the PSB-VVER Installation", *Int. Topical Meet. on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon (F), October 2-6*.

D'Auria F. and Galassi G.M., 1990, "Characterization of Instabilities during Two-Phase Natural Circulation in PWR typical conditions", *J. Experimental Thermal and Fluid Science*, ***Vol. 3***.

D'Auria F. and Galassi G.M., 1990a, "Flow-rate and Density Oscillations during Two-Phase Natural Circulation in PWR typical Conditions", *J. Nuclear Engineering and Design,* ***Vol 122***.

D'Auria F., Galassi G.M., Vigni P., Calastri A., 1991, "Scaling of Natural Circulation in PWR Systems", *J. Nuclear Engineering and Design*, ***Vol 132***, *pages 187-205*.

D'Auria F., Galassi G.M., Ingegneri M., 1994, "Evaluation of the data base from high power and low power small break LOCA counterpart tests performed in LOBI, SPES, BETHSY and LSTF facilities", *University of Pisa Report, DCMN - NT 237(94), Pisa (I), November*.

D'Auria F., Frogheri M., Leonardi M., 1994a, "Natural Circulation Performance in Western type and Eastern type PWR", *Simulator Multiconference, San Diego (CA), April 11-13*.

D'Auria F., Frogheri M., Vigni P., 1995, "Natural Circulation in CANDU, WWER and PWR type Nuclear Plants", 13th Conf. of Italian Society of Heat Transport, Bologna (I), June 22-23.

D'Auria F., Galassi G.M., Frogheri M, 1998, "Natural Circulation Performance in Nuclear Power Plants, *2nd Conf. of the Croatian Nuclear Society, Dubrovnik (HR), June 17-19*.

D'Auria F. and Frogheri M., 2002, "Use of Natural circulation map for assessing PWR performance", *J. Nuclear Engineering & Design (Special Issue),* ***Vol 215***, *Nrs 1&2, pages 111-126*.

D'Auria F. and Galassi G.M., 2002, "Final Results from the analysis by Relap5 code of the RD-14m test B-9401", *University of Pisa Report, DIMNP - NT 471(02), Pisa (I), July 2002, IAEA Advisory Group on Inter-comparison and Validation of Computer Codes for Thermal-hydraulic Safety Analyses of Heavy Water Reactors, Vienna (A)*.

D'Auria F., Cherubini M, Galassi G.M, Giannotti W., Ferreri J.C., Mazzantini O., 2006, "Use Of The Natural Circulation Flow Map", *IAEA 3rd CRP Meeting, Natural Circulation In Water-Cooled Nuclear Power Plants, Cadarache, France, Sept. 10-15.*

D'Auria F., Del Nevo A., Muellner N., 2007, "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, June 23-27, Paper ID. T07.*

D'Auria F. and Galassi G.M., 2007, "Natural Circulation situations relevant to Nuclear Power Plants", *IAEA Course on Natural Circulation in Water-Cooled Nuclear Power Plants, International Centre for Theoretical Physics (ICTP), Trieste, Italy, June 23-27, Paper ID. TXX.*

Del Nevo A., D'Auria F., Galassi G.M., 2006, "Cathare2v1.5b post test analysis and accuracy quantification of PKL-III NC test F1.2 (restricted)", *University of Pisa Report, DIMNP - NT 559(06)-rev0, Pisa (I)*.

Ferreri J.C., Mazzantini O., Ventura M.A., Rosso R.D., D'Auria F., 2003, "Natural Circulation in the CNA-I PHWR NPP", *10th Int. Top. Meet. on Nuclear Reactor Thermal-hydraulics (NURETH-10), Seoul (Korea), Oct. 5-9*.

IAEA, 2002, "Natural Circulation data and methods for advanced water cooled nuclear power plants design", *IAEA Tecdoc 1281, Vienna (A),*

IAEA, 2005, "Natural Circulation in water cooled nuclear power plants", *IAEA Tecdoc 1474, Vienna (A).*

IAEA, 2006, "Status of innovative small and medium sized reactor designs 2005", *IAEA Tecdoc 1485, Vienna (A)*.

Kukita Y. and Tasaka K., 1989, "Single Phase Natural Circulation in Pressurized Water Reactors under degraded Secondary Cooling Conditions", *ASME Winter Meeting.*

Mousavian S.K., D'Auria F., Salehi M.A., 2004, "Analysis of natural circulation phenomena in VVER-1000", *J. Nuclear Engineering & Design, **Vol 229**, pages 25-46*.

Prosek A., Mavko B., D'Auria F., 2004, "Quantitative analysis of RD-14M large LOCA test B9401 calculations", *IJS Report IJS-DP-8819, Ljubljana (Slo)*.

Richards D. J., D'Auria F., Bedrossian G.., Cho Y. J., Gupta S. K., Istrate R., 2003, "Inter-comparison and validation of computer codes for thermal-hydraulic safety analyses of heavy water reactors", *10th Int. Top. Meet. on Nuclear Reactor Thermal-hydraulics (NURETH-10), Seoul (Korea), Oct. 5-9*.

Tuunanen J., Vihavainen J., D'Auria F., Frogheri M., Galassi G.M., Kimber G., Lillington J., Allen E., Williams T.G.., 2000, "Analyses of PACTEL passive safety injection experiments with APROS, CATHARE and RELAP5 codes", *J. Nuclear Engineering & Design, **Vol 198**, pages 261-273*.