Natural circulation systems in nuclear reactors: advantages and challenges

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Abstract. Following the Fukushima disaster in Japan in March 2011, a number of questions and doubts about the safety of nuclear reactors have been raised globally and many nations have begun revising and improving the nuclear installations. Natural circulation plays a major role in nuclear accident. This significance of the natural circulation-based systems led us to study the natural circulation phenomena for better understanding and in order to ensure the safety and efficiency of natural circulation systems. This review focuses on the advantages and challenges of natural circulation systems.

1. Introduction

Over the previous fifty years, nuclear power has become a significant component of the energy mix in many nations. Ninety percent of the world's present fleet are watercooled reactors and the majority of new nuclear power plants (NPPs) are expected to be advanced water-cooled reactors in the future[1]. In several industrial systems, natural circulation (NC) is a significant mechanism and understanding of its operation is of concern in the design, operation and safety of nuclear reactors.

The NC is a heat transfer process which driven by the difference of the densities under the force of gravity[2]. The natural circulation mode is a process that coolant flows in a closed loop without any effects of external force[3]. When the commercial use of natural circulation systems (NCSs) started as heat transport systems, it is hard to identify. In the automotive industry, the first large-scale use of these technologies appears to have cooled the engine block. With the introduction of high compression internal combustion engines their utilization in the automotive industry nearly ceased in the 1940s[4]. Therefore, at reactor accidents in PWR, NC is widely used as an energy transfer system to remove the residual heat from the reactor core. Some new advanced power plant designs depend on active systems with high safety criteria and others are provided by passive systems of high reliability to meet safety requirements and reduce the cost[4].

For the safety of nuclear reactor, the NC phenomenon of nuclear reactors, the NC phenomenon is very important. Advanced reactors were intended using NC-based passive safety systems[5]. AP 600 is an example of a new generation of PWR which provided by passive safety systems to implement the safety feature when severe accident such as loss of flow or LOCA, happens[6]. The Passive safety systems employ NC as process to remove the residual heat from the core to the outside which emphasize the importance of the phenomena in nuclear reactor design. So NC is one of the most important physical phenomena describing the thermal-hydraulic attitudes of the Passive safety systems incorporated in a new generation of PWR[7].
After Fukushima disaster on March 2011 in Japan, several queries and doubts associated with the safety of nuclear reactors have raised worldwide[8]. The accident has disclosed several deficiencies in presently operative nuclear energy plants that should be resolved to avoid within the future similar issues. Many lessons have been learned from the accident, so it was necessary for researchers, engineers and scientists to continue reviewing Safety protocols, work on safer reactor design and conduct safety analysis and experiments. This review is another step on the road towards a safest nuclear future, which summaries and reflects the important role of NC in nuclear safety as well as highlights the important NC experiments, advantages, and challenges.

2. Advantages and challenges of NCSs

Compared to forced flow schemes, NCSs have their own advantages and disadvantages in both areas of plant safety and plant economy. A NCSs main advantage is simplicity, the removal of active power supplies and pumps can significantly simplify the system's constructions operation and maintenance due to its simplified configuration.

In addition, the exclusion of pumps and connecting piping also excludes accident scenarios connected with loss of pump flow, pump seal rupture and loop seal manometer impacts during Small Break Loss-of-Coolant-Accidents (SBLOCAs). Another advantage is that in a NC scheme, the flow distribution in parallel channel cores is much more uniform. With the assumption that the pressure in the inlet is uniform so the pressure drop is constant in the parallel channels. Moreover, in a NC scheme, the two-phase fluid flow features as a function of energy are also better. That is, the flow rises with power, while the flow reduces with an increase in power in a forced two-phase fluid scheme. However, in comparison with forced flow schemes of the same power rating, NC reactors systems tend to have large volumes and comparatively low power densities. As a consequence, the NCSs thermal response is slow, allowing operators sufficient time to react to plant upsets[4].

A NCSs main disadvantage is that the driving force is small. The only way to achieve fairly high flow rates with low driving power is to design for low pressure losses. Compared to forced circulation schemes, low driving force and the consequent use of big diameter components lead to low mass flux in NC systems. Flow instability is the most important disadvantage for NCSs and it is very challenging for scientists and researchers and of utmost importance for industrial applications[9]. While instability is common to NCSs and forced circulation systems, the NCSs are inherently less stable than forced circulation systems. Several decades have been spent studying flow instability and, in the past, several investigations have been conducted in this field numerically and experimentally. NCSs are vulnerable to various types of instabilities[10]. Figure 1 presents the summary of the classification of the flow instabilities as discussed by [9] and [11].

Figure 1. Types of flow instabilities (reprinted from [9]).
Numerous researches have been conducted to study the flow instabilities in a NC boiling systems and boiling water reactors [12], [13], [14], [15], [16]. Also, different types of instabilities have been analyzed in the literature for supercritical flow and supercritical pressure in a NC loop [17], [18], [19]. In all these papers that discussed the flow instabilities, the scholars indicated the importance of NC instabilities, as a result, their focus is on the development of new models and providing challenging data that can be used for validation of different computer codes.

3. Classification of NCSs
NCSs can be categorized according to the phase of the coolant and depending on the thermodynamic state of the working fluid, NCSs are classified as single-phase, two-phase and supercritical NCSs.

3.1. Single-phase NCSs
In a single-phase NCSs, the circulating fluid in the entire loop continues to remain in only one state. Many works were dedicated in the literature to investigate the single-phase NCSs, Swapnalee et al [20] reported generalized flow equation relevant to natural single-phase circulation that is valid only in instances where the whole loop follows a single friction law. The suggested equation is tested in a uniform rectangular loop with experimental information produced and is discovered to be in excellent agreement. Also, an old study was carried out By ALSTAD et al [21], the study illustrated a method for predicting the transient behavior of a single-phase NC loop. The method uses the iterative solution of the energy and momentum balance of the finite difference and has many points of concern. Tian et al [22] performed experimental study of single-phase, NC heat transfer characteristics in a narrow, vertical, rectangular channel under static and rolling motion conditions; The experimental findings indicate that in rolling motion the NC flow rate differs periodically. In addition, rolling motion leads to cycle-averaged flow rate decrement. Wu et al [23] studied an innovative flow-resistance performance in the single-phase NC loop theoretically and experimentally. Taking into account the impacts of local and frictional resistance in the NC loop, the analytical relationship between the NC mass flow rate and the heating section temperature rise with heating power, are all acquired through the approximation and fitting approach. The Study of NC phenomena during a single-phase loop with unified diameters was performed by Vijayan [24]. Turbulent and laminar flows were applied in this work and the effect Grashof, Reynolds number and mass flow were observed. The agreement of experimental results with the correlations was satisfied. Rao et al. [25] proposed dynamic performance under step, ramp, exponential and sinusoidal excitations in NC loop. However, Luzzi et al. [26] present a semi-analytical and numerical model to study the dynamic behavior of NC loop. The model match with the experimental data. Misale [27] study the thermohydraulic performance of a NC loop under constant and variable power. All the test showed unstable behavior. A mathematical model proposed by Mousavian et al. [27] and compared with RELAP5 code to study single phase NC.

3.2. Two-phase NCSs
Two-phase NCSs either with only boiling or with both boiling and condensation are relevant to NPPs. Commonly, two-phase NC occurs in a BWR following the failure of circulating pumps. In PWRs, PHWRs or VVERs it is possible to be observed following the partial loss of coolant inventory in case of a small break LOCA with pump failure. Many scholars discussed the two-flow NC phenomena as an important point on the recent issues related to nuclear power plant [27], [28], [29], [30], [31] mainly their focus goes to the stability of the two-phase flow and the different types of instabilities. Gartia et al [32] proposed a generalized flow correlation to estimate the steady-state flow in two-phase NC loops. Many research conducted to explore the capability of different computer codes as well as mathematical models for prediction of two-phase flow NC behavior, the capabilities of RELAP5 code is studied in [33-38]. Figure 2 depicted typical comparison of the measured flow rates with the different code predictions. RELAP5 code gives lower predictions (5–8%) than the test data while predictions of theoretical and TINFLO are, 5–7% and 5–13% higher respectively. Several outstanding
experimental and analytical research to study flow characteristics in two phase NC loop have been conducted such as Filho et al.[39], Yan et al.[40], Zhang et al.[41] and Ruspini et al.[42].

3.3. Supercritical NCSs

The primary interest in supercritical structures arises from the reality that there is a big shift in the volumetric coefficient of thermal expansion near the critical point and they are therefore able to generate driving forces similar to those of two-phase NCSs.

Another feature is that supercritical water systems have good heat transfer characteristics [4],[43].

Many articles state the fact that supercritical carbondioxide can be used as a working fluid of high efficiency in NC such as [44],[45]. Cao et al.[46] proposed a two dimensional numerical model to study the convective flow and heat transfer characteristics of supercritical CO$_2$ NC in a uniform diameter rectangular loop. Also, Chen et al.[47] conducted numerical simulations on a supercritical CO$_2$ NC loop to analyse the transitions in flow and instability of such systems. Another research done by Liu et al.[48] to study the effect of buoyancy and flow acceleration on the heat transfer behavior of supercritical CO$_2$ in the heating section in a NC loop.

![Figure 2. Comparison of code prediction with experimental data (reprinted from Gartia et al[32]).](image)

4. NC Experiments

The advantages of using NC as a means of core heat removal has prompted the worldwide development of separate effects and integral system test facilities. Various NC experiments and investigations are described in the literature. Jang et al.[49] evaluate the NC capability of REX-10 test facility. NC studies are conducted with different heater powers, pressures, and secondary flow rates. The findings of the studies are that the power have a major impact on the NC flow. An experimental study conducted by Hong-bo and Dong-Hua[50] on the multipurpose supercritical water test loop which contain 2×2 bundle. In 2×2 bundle, two types of heat transfer deterioration are noted, and the first type of heat transfer two type deterioration with reduced mass flux and greater heat flux is more probable to happen.

Another Experimental and numerical investigations of NC phenomena in passive safety systems for decay heat removal in large pools conducted by Krepper and Beyer[51]. They analyze the ability of current CFD codes to describe these phenomena. Jain et al.[52] studied Experimentally the flow instability behavior of a multi-channel boiling NC loop, they discussed the evolution of unstable and stable behavior, together with the nature of the channel flow oscillation and the pressure impact on it. Vyas et al.[53] reported results of experimental research on the features of a two-phase natural low pressure circulation system with parallel boiling channels having their own heat sources. The NC flow was noted to be fluctuating through parallel boiling channels. It was noted that with growing
downcomer resistance, the two-phase NC flow does not always decrease. Lemos et al.[54] presented an experimental study about fluid flows behavior in NC, under conditions of single-phase flow. The experiment contributed positively to the acquisition of significant data on the NC phenomenon and to the identification of true operating conditions in the shutdown or emergency of nuclear power plants from the studies conducted under comparable conditions. Kun et al.[55] performed an experimental analysis in a closed NC loop to evaluate the characteristics of low-pressure pressure drop oscillations (PDO). The most important findings are the PDO with large amplitude thermal-hydraulic oscillations and reverse flow may occur in a closed NC loop when the pressurizer is connected to the upstream section of the test.

All these experiments and investigations were performed on different scaled test facilities at different operating conditions and with different arrangements. However, their outcomes are applicable only to those configurations and operating conditions, which are similar to conditions for which they were derived.

5. NC for small break loss of coolant accident (LOCA)

Following the accident on Three Mile Island in 1979, many studies on the safety of a nuclear reactor during a SBLOCA were performed to better understand physical phenomena and to develop quantitative methods for simulating thermal hydraulic behavior in a reactor coolant system during the break[56]. Up to date, numerous integral effect test facilities have been constructed and used to support safety analysis for nuclear power plants[57]. The information from these facilities were used to define a broad range of thermal hydraulic phenomenon that are essential for NCSs, as well as to evaluate the predictive capacities of a variety of thermal hydraulic evaluation codes. Some example of test facilities are LSTF (Large Scale Test Facility) in Japan[58], [59], ATLAS (Advanced Thermal-hydraulic test Loop for Accident Simulation) in Korea[60],[61], advanced core-cooling mechanism experiment (ACME) in China [62], and PANDA facility built by the Paul Scherrer Institute (PSI)[63].

Wang et al. pointed out that although much work was done, the dominant mechanisms of the transient SBLOCA and the significant thermal hydraulic behaviors were not clearly understood due to the complexity of the passive safety geometry of the nuclear reactor combined with the transient two-phase fluid interactions [57], [64]. The transient SBLOCA in AP1000 can be segregated into four different phases: the blow-down phase, the NC phase, the blow-down phase of the ADS and the injection phase of the IRWST. A typical pressure transient for an AP600/AP1000 SBLOCA is shown in Figure 3. Different thermal hydraulic phenomena characterize each phase of AP1000 SBLOCA chronology.

![Figure 3. AP1000 small break LOCA pressure transient (reprinted from Wang et al.[57]).](image-url)
Takeda et al.[58] simulated a PWR 1 percent cold leg small break LOCA with the premise of high-power NC owing to scram failure and complete high-pressure injection system failure using RELAP5/MOD3.2.1.2 code. The RELAP5 code anticipated the general thermal-hydraulic phenomena observed in the experiment fairly well.

A countless number of paper and research debated on SBLOCA in PWR [65], [66]. Because NC is an important key heat rejection mechanism, a thorough understanding of NC mechanisms and variables that affect the NC reaction of the reactor system is necessary. However, in severe accidents, the importance of NC flow is that it transfers energy from the core to the other RCS regions.

6. Summary
A large number of numerical and experimental investigations on NCSs have been reported in the literature. However, an outstanding experimental database covering distinct operating ranges, i.e. low-power low-pressure (start-up condition), high-power high-pressure (normal condition) and low-power high-pressure (shutdown condition) as well as various transient conditions such as SBLOCA was established. Many experimental and numerical investigations have been done to classify and study the NC flow instability. It was classified to static and dynamic instabilities. And the application of Al2O3 nanofluids in NCSs improve the NC flow rate and suppressed the flow instabilities. The current and future work goes to the new generation BWRs which, have NC as the normal operation mode. In order to visualize NC instability phenomena under broad operating conditions, more extensive experimental investigation is required. In designing future BWRs, the knowledge thus obtained can be used.

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References
[1] F Castiglia, M Giardina 2010 Nucl. Eng. Des. 240 2779-2788
[2] Seyed Khalil Mousavian, Francesco D’Auria, Mahmoud A. Salehi 2004 Nucl. Eng. Des. 229 25-46
[3] Haojie Cheng, Haiyan Lei, Long Zeng, Chuanshan Dai 2019 Int. J. Heat Mass Transf. 128 208-216
[4] TECDOC-1474 IAEA, 2005
[5] Dilip Saha, John Cleveland 2008 Sci. Technol. Nucl. Install.
[6] Donghua Lu, Zejun Xiao, Bingde Chen 2010 Ann. Nucl. Energy 37 691-700
[7] TECDOC-1624 IAEA, 2009
[8] C Stan-Sion 2019 Nucl. Instruments Methods Phys. Res. Sect. B Beam Interact. with Mater. Atoms 438 107-112
[9] Gonella V Durga Prasad, Mannmohan Pandey, Manjeet S Kalra 2007 Prog. Nucl. Energy 49 429-451
[10] A K Nayak, P K Vijayan 2008 Sci. Technol. Nucl. Install. 2008
[11] JoséM Rey José March-Leuba 1993 Nucl. Eng. Des. 145 97-111
[12] Kazem Arrdaneh, Salman Zaferanlouei, Mohsen Farahi, Asad Ahmadi 2010 in of the 14th International Heat Transfer Conference IHTC14 1-10
[13] Gonella V Durga Prasad, Mannmohan Pandey, Santosh K Pradhan, Satish K Gupta 2008 Nucl. Eng. Des. 238 1750-1761
[14] G V Durga Prasad, Mannmohan Pandey 2008 Nucl. Eng. Des. 238 229-240
[15] C P Marcel, M Rohde, T H J Van der Hagen 2008 Int. J. Heat Mass Transf. 51 566–575
[16] A K Nayak, M R Gartia, P K Vijayan 2008 Exp. Therm. Fluid Sci. 33 184-189
[17] B. T. Swapnalee, P. K. Vijayan, M. Sharma, D. S. Pilkhwal, Nucl. Eng. Des., 245, 99–112, 2012.
[18] Jiyang Yu, Shuwei Che, Ran Li, Bingxue Qi 2011 Prog. Nucl. Energy 53 775-779
[19] Prashant K Jain, Rizwan-uddin 2008 Nucl. Eng. Des. 238 1947-1957
[20] B T Swapnalee, P K Vijayan 2011 *Int. J. Heat Mass Transf.* 54 2618-2629
[21] C D Alstad, H S Isbin, N R Amundson, J P Silvers 1955 *AIChE J.* 1 417-425
[22] Wangsheng Tian, Xiaxin Cao, Changqi Yan, Zhenxing Wu 2017 *Int. J. Heat Mass Transf.* 107 592-606
[23] Lei Wu, Yang Liu, Hai jun Jia, Jun Wang 2017 *Int. J. Heat Mass Transf.* 107 66-73
[24] P K Vijayan 2002 *Nucl. Eng. Des.* 215 139-152
[25] N M Rao, B Maiti, P K Das 2005 *Int. J. Heat Mass Transf.* 48 3185-3196
[26] L Luzzi, M Misale, F Devia, A Pini, M T Cauzzi, F Fanale, A Cammi 2017 *Chem. Eng. Sci.* 162 262-283
[27] M Misale 2016 *Int. J. Heat Mass Transf.* 99 782-791
[28] R N De Mesquita, P H F Masotti, R M L Penha, D A Andrade, G Sabundjian, W M Torres, L A MacEdo 2012 *Nucl. Eng. Des.* 250 592-599
[29] P K Vijayan, A K Nayak, D Saha, M R Gartia 2008 *Sci. Technol. Nucl. Install.* 2008
[30] Zhiee Jhia Ooi, Vineet Kumar, Caleb S 2019 Brooks *Progress in Nuclear Energy* 116 124-136
[31] P T Senda, R T Dobson 2018 *R D J. if South African Inst. Mech. Eng.* 61-71
[32] M R Gartia, P K Vijayan, D S Pilkhwal 2006 *Nucl. Eng. Des.* 236 1800-1809
[33] Zhao Guozhi, Cao Xinrong, Shi Xingwei 2013 *Ann. Nucl. Energy* 60 115-126
[34] Qiang Wang, Puzhen Gao, Xianbing Chen, Zhongyi Wang, Ying Huang 2018 *Ann. Nucl. Energy* 121 210-222
[35] Daniele Martelli, Nicola Forgione, Gianluca Barone, Ivan di Piazza 2017 *Ann. Nucl. Energy* 101 408-418
[36] Amit Mangal, Vikas Jain, A K Nayak 2012 *Prog. Nucl. Energy* 61 1-16
[37] M M Stempniewicz, M LF Slootman, H T Wiersema 2016 *Nucl. Eng. Des.* 307 130-143
[38] Y Kozmenkov, U Rohde, A Manera 2012 *Nucl. Eng. Des.* 243 168-175
[39] Francisco A.Braz Filho, Gaianê Sabundjian, Guilherme B. Ribeiro, Alexandre D. Caldeira 2017 *Ann. Nucl. Energy* 105 249-258
[40] Xiao Yan, Guangming Fan, Zhongning Sun 2017 *Ann. Nucl. Energy* 104 291-300
[41] Y J Zhang, G H Su, X B Yang, S Z Qiu 2009 *Nucl. Eng. Des.* 239 1294-1303
[42] Leonardo Carlos Ruspini, Christian Pablo Marcel, Alejandro Clausse 2014 *Int. J. Heat Mass Transf.* 71 521-548
[43] Attila Kiss, Mártón Balaskó, László Horváth, Zoltán Kis, Attila Aszódi 2017 in *Annals of Nuclear Energy* 100 178-203
[44] Guangxu Liu, Yanping Huang, Junfeng Wang, Fa Lv, Laurence K.H. Leung 2016 *Nucl. Eng. Des.* 300 376-383
[45] Milan Krishna Singh Sarkar, Dipankar Narayan Basu2017 *Nucl. Eng. Technol.* 49 103-112
[46] Yuhui Cao, Xin Rong Zhang 2012 *Int. J. Therm. Sci.* 58 52-60
[47] Lin Chen, Xin Rong Zhang, Hiroshi Yamaguchi, Zhong Sheng Liu 2010 *Int. J. Heat Mass Transf.* 53 4101-4111
[48] Guangxu Liu, Yanping Huang, Junfeng Wang, Fa Lv 2015 *Int. J. Heat Mass Transf.* 91 640-646
[49] Byeong-ill Jang, Hyeong-min Joo, Sun-do Choi, Gyoo-dong Jeun, Moo-hwan Kim 2009 *Mech. Eng.* 667-668
[50] L I Hong-bo, L U Dong-hua 2017 in *Proceedings of the 2017 25th International Conference on Nuclear Engineering ICONE25* 1-9
[51] Eckhard Krepper, Matthias Beyer 2010 *Nucl. Eng. Des.* 240 3170-3177
[52] Vikas Jain, A K Nayak, P K Vijayan, D Saha, R K Sinha 2010 *Exp. Therm. Fluid Sci.* 34 776-787
[53] H P Vyas, V Venkat Raj, A K Nayak 2010 *Nucl. Eng. Des.* 240 3862-3867
[54] Wanderley Freitas Lemos, Nuclear Engineering Program, Rio De Janeiro, Jian Su, in 2011 *International Nuclear Atlantic Conference - INAC 2011*, 2011
[55] Kun Cheng, Tao Meng, Chunping Tian, Hongsheng Yuan, Sichao Tan 2018 *Int. J. Heat Mass Transf.* 122 1162-1171
[56] Kouhei Kawanishi, Ayao Tsuge, Makoto Fujiwara, Tamio Kohriyama, Hiroichi Nagumo 1991 *J. Nucl. Sci. Technol.* **28** 555-569
[57] W W Wang, G H Su, S Z Qiu, W X Tian 2011 *Prog. Nucl. Energy* **53** 407-419
[58] Takeshi Takeda, Hideaki Asaka, Hideo Nakamura 2009 *Ann. Nucl. Energy* **36** 386-392
[59] Hideo Nakamura, Tadashi Watanabe, Takeshi Takeda, Yu Maruyama, Mitsuhiro Suzuki 2009 *Nucl. Eng. Technol.* **41** 753-764
[60] Chul Hwa Song, Ki Yong Choi, Kyoung Ho Kang 2015 *Nucl. Eng. Des.* **294** 242-261
[61] Yeon Sik Kim, Ki Yong Choi, Chul Hwa Song, Won Pil Baek 2014 *Ann. Nucl. Energy* **63** 509-524
[62] Yu Quan Li, Hua Jian Chang, Zi Shen Ye, Fang Fang Fang, Yan Shi, Kai Yang, Ming Tao Cui 2016 *Prog. Nucl. Energy* **88** 375-397
[63] Domenico Paladino, Jörg Dreier, *Sci. Technol. Nucl. Install.*, 2012, 2012.
[64] W W Wang, G H Su, W X Tian, S Z Qiu 2013 *Nucl. Eng. Des.* **263** 380-394
[65] Takeshi Takeda, Akihiko Ohwada, Hideo Nakamura 2013 *Exp. Therm. Fluid Sci.* **51** 112-121
[66] Taisuke Yonomoto, Masaya Kondo, Yutaka Kukita 1997 *J. Nucl. Sci. Technol.* **34** 571-581