Nuclear Energy in Central Europe '98
Terme Čatež, September 7 to 10, 1998

A potential of boiling water power reactors with a natural circulation of a coolant

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Abstract

The use of the natural circulation of coolant in the boiling water reactors simplifies a reactor control and facilitates the service of the equipment components. The moderated core power loads allow the long fuel burnup, good controll ability and large water stock set up the enhancement of safety level. That is considered to be very important for isolated regions or small countries.

In the paper a high safety level and effectiveness of BWRs with natural circulation are reviewed. The limitations of flow stability and protection measures are being discussed. Some recent efforts in designing of such reactors are described.

1. Introduction

On the end of the 20th century the nuclear community is being faced with unexpected difficulties in the spreading of the nuclear energy. A gas industry expansion and some known accidents at NPPs have influenced on the progress in this area of power engineering. A situation in a development of the NPPs could be called a stagnation. Only in the far east there are remarkable activities on a construction of NPPs.

There is no doubt among the knowledgeable experts, that in long-term expectations nuclear power plants are sufficiently safe, economically effective and can be considered as a sustainable power supply. In Europe the "gas pause" could be used for detailed analyse of the alternatives and for choosing the proper reactor type for national energy supply.

The dates of latest design activities show that the Boiling Water Reactors with natural circulation of the coolant (BWR-NC) are reliable and economically attractive power sources. It could be understood that in the state with the permanent budget deficit the BWR-NC could be considered as possible basis for nuclear power stations. It is thought that such reactors are the best choice among the various alternatives to get proper safety level of NPP with the moderate expenses.

2. Boiling Water Reactors in the world

For a long time many countries are using the boiling water reactors for energy production. Comparing to the PWRs have such reactors lower operating pressure in the vessel, lower number of heavy components, smaller volumes of the technological premises and consequently, less capital and operational expenses.

In the beginning of 1997 there were 94 BWRs in operation among 433 nuclear power units. A successful operation of BWRs has proved, that the principle of the direct cycle did not create significant difficulties for services of the equipment, while the stability and vapor radioactivity problems are being overestimated. All these features allow constructing the BWRs with electrical power up to 1350 MWe. ABWRs successfully operate in Japan [1]. It is appropriate to mention some modern activities in Germany to design a reactor SWR 1000 [2].

Among the measures to improve the safety the recommendations considering the simplification of the systems occupy the important place. The use of natural circulation of the water in the reactor vessel creates a new level of the reactor control simplification. It also facilitates the service of the equipment components and enhances the safety of the plant. The moderate core power loads allow the long fuel burnup. A good controll ability and large water stock set up the improvement of safety level. That is considered as a very important feature of NPPs for isolated regions or small countries. Current information shows a definite interest of the reactor designers for BWRs with natural circulation of the coolant. It is well known, that since 1982 a Simplified Boiling Water Reactor (SBWR) with electric capacity 670 MWe is being developed by "General Electric" in USA [3].

The SBWR makes extensive use of passive features and simplified systems to produce cost-saving advantages. Key design features of SBWR includes the systems for depressurization and pressure suppression, gravity driven cooling system and other features, that rely on gravity or stored energy to ensure core cooling, decay heat removal and ATWS mitigation.

On the International Conference TOPNUX-96, Paris, France this company has presented very detailed reports about the project of the European Simplified Boiling Water Reactor (ESBWR) [4], which was designed to meet the requirements for new NPPs to be built in Europe. The main design features of that ESBWR 1190 MWe (net) plant are as follows:

a) Simplicity achieved by use of natural circulation and passive safety systems,
b) Improved plant economics - by reduction of systems and buildings, by use of reliable systems and standard components from operating BWRs,
c) Enhanced safety - by passive safety and diverse non-safety systems.

The technical characteristics of SBWR, ESBWR and ABWR are summarized in Table 1.

Table 1: Technical Data of reactors

Plants	SBWR	ESBWR	ABWR
Power [MWth]	2000	3613	3926
Net Electrical Output [MWe]	670	1190	1356
Vessel-diameter,	6,0	7,1	7,1
height [m]	24,5	24,9	21,1
Power density [kW/m]	41,5	48	51
Core height [m]	2,74	2,74	3,71
Equivalent diameter [m]	4,73	5,9	5,4
No. control rods drives	177	269	205
Circulation	Natural	Natural	Forced
No. of Bundles	732	1132	872
Vessel pressure, [MPa]	7,17	7,17	7,17

In Russia a development of the BWRs led to the construction of the RBMK reactors. A Chernobyl accident stopped their story. The only Boiling Light- Water moderated and cooled reactor which is being successfully operated since 1965, is the VK-50, located in Dimitrovgrad, Ulianovsk region [5].

A similar boiling reactor with NC was a Dodewaard reactor in Netherlands that has been constructed roughly same time [6], however it is now closed after 29 years of successful operation.

Broad experimental activities in various areas of reactor technology, good result gathered during the operation of these and other reactors with direct cycle have proved the prospects of BWR-NC types. Recently the development of reactor VK-300 has been initiated in Russia [7].

Table 2: Technical Data of some reactors with natural circulation

Plants	VK-50	Dodewaard	VK-300
Power [MWth]	250	163,4	900
Net Electrical Output [MWe]	50	50	300
Vessel- diameter [m]	3,8	2,794	4,58
Height [m]	11,2	12,09	12,4
Power density [MW/l]	41,5	36,3	28,0
Core height [m]	2,0	1,793	3,5
Equivalent diameter [m]	2,8	1,788	3,4
No. of control rod drives	31	37	90
Circulation	Natural	Natural	Natural
No. of Bundles	187	156	151
Vessel pressure [Mpa]	10,0	7,15	7,0

3. Features of the BWRs with natural circulation

3.1 Stability

A problem of the nuclear-coupled instability has been a major concern since the early stage of BWR development. In US Code of Federal Regulations (10 CFR 50, Appendix A) among criteria for protection core there is Criterion 12- "Suppression of reactor power oscillations", which has to be met during design and operation of NPPs.

An extensive program of experiments has been carried out in USA and other countries to disclose the nature of that phenomenon. Nevertheless in some cases the real instability of events has been observed under the operation of NPPs (for example, power oscillations at LaSalle NPP in March 1988 [8]).

It was discovered that there are neutronic, thermo-hydraulic and acoustic types of the instabilities, which resulted from lags in the feedback links between power, flowrate and liquid density. Those instabilities influenced on reactivity between the driving forces and flowrate. Indeed, the coolant density oscillations are creating the local disturbances in neutron multiplication and driving forces, which influenced delayed power change and flowrates. The consequence is the change of coolant density.

There are two possibilities for analyzing the unstability phenomenon- the numerical investigation of the system models on the digital computers and semi-analytical consideration of simplified models. In the latter case a linear stability analysis is being used to determine the threshold of oscillatory behavior of the boiling reactor [9, 10].

In linear analysis interactions of the parameters are described in the terms of transfer functions. In this case transfer function describes the response of the reactor power as a function of small harmonic disturbances of the external reactivity. It was shown that the neutron dynamics of the reactors with feedback could be described by differential equations of point neutronic kinetic,

$$l\frac{dN}{dt} = (\rho - \beta)N + \sum_i \lambda_i C_i \qquad \frac{dC_i}{dt} = -\lambda_i C_i + \beta_i N,$$

with reactivity calculated as

$$\rho(t) = \rho_{in} - \rho_{fb}\ , \quad \rho_{fb} = \Sigma_k\ \rho_{fbkd}$$

$$\rho_{fbk} = \frac{\partial\rho}{\partial f_k} f_k \int_{-\infty}^{t} N(t')F_k(t-t')dt',$$

where are:

ρ_{in} - external disturbances of the reactivity,
ρ_{fbk} - feedback reactivity from k-th parameter (temperature of the fuel, average void fraction and others),
$F_k(t)=f_k(t,N)$- time response of parameter f_k on power changes.

After linearization and Laplace-transformation, this system can be presented as an algebraical equation

$$s\,[l + \Sigma_i\ \beta_i/(s+\lambda_i)]\delta N^*/N = \delta\rho_{in}^* - \Sigma_k\,(\partial\rho/\partial f_k)\ f_k\, F^*_k(s)\ \delta N^*/N\ , \quad \text{where}$$

$$\delta N^*(s) = \int_o^{\infty} \delta N(t)\exp(-st)dt \quad \text{or} \quad \{Do^{-1} + \Sigma_k\,(\partial\rho/\partial f_k)f_k F_k^*\ \}\delta N^*/N = \delta\rho_{in}^*,$$

where is $D_o = \{s[l + \Sigma_i\ \beta_i/(s+\lambda_i)]\}^{-1}$ a transfer function of the reactor with zero power.

The transfer functions of the reactor with negative void-fraction feedback can be presented by (with $s=i\omega$)

$$\frac{\delta N^*}{N\,\delta\rho_{in}^*} = D(i\omega) = \frac{D_o(i\omega)}{1 + D_o(i\omega)\dfrac{\partial\rho}{\partial\varphi}\varphi F^*(i\omega)}\ ,$$

where is $F^*=(\delta\varphi^*/\ \varphi)[\delta N^*/N]^{-1}$ a transfer function of the relative changes of the void fraction at changes of the power.

Usually function F^* can be approximated by chain of three links. The condition of escaping from resonant instability can be approximatelly presented as

$$\frac{K}{(1+\tau_f s)(1+\tau_s s)(1+\tau_b s)} \le -1\,,$$

where are:

$K=(1/\beta)(\partial\rho/\partial\varphi)\ \varphi$ - vapor reactivity,
τ_f - time constant of the fuel element,
τ_s, τ_b - subcooled and boiling transport times.

It is possible to show by using Routh-Hurwitz criterion, that a system will be stable, if

$$K<(1+\frac{\tau_s}{\tau_f}+\frac{\tau_b}{\tau_f})(1+\frac{\tau_f}{\tau_s}+\frac{\tau_f}{\tau_b})-1 .$$

A detailed analysis of neutronic unstability is being performed on the computers. An effective code for such actions has been developed in Kurchatov Institute [11].

In the VK-50 reactor an unstability threshold is determined as a point of vanishing of the decrement of the noise correlation function.

Another problem is hydraulic instabillity of the natural circulation. Instability of flowrate can be static and dynamic. It is known as "chugging" behavior of steam boilers, thermosyphons and other two- phase flow system with time lags between vapor production and flow rates. Theoretical analysis of such phenomena is presented in [9,12]. Recently a great attention is being paid for investigation of the nonlinear dynamics of the BWRs [13].

3.2 Radiation in the compartments.

Specific feature of the direct cycle is using the steam, generated in the core, which get some radioactivity of the oxygen. It is carrying fission products from fuel elements and some corrosion products. A well-known measures like high quality of the fuel, cleaning up of the coolant and degassing treatment of the steam, provide the acceptable radiation levels in working halls of the NPPs. Radiation levels in some rooms of the VK-50 are presented in the Table 3.

Table 3: Intensity of the radiation near some equipment of the reactor VK-50

Equipment	Intensity of radiation, during operation	[mrem/h] on shut down
Pipeline on reactor outlet	750-1000	9 - 13
Stop valve of turbine	35- 40	1,8- 4,4
Low pressure cylinder	10- 15	0,4- 0,8
Condenser of turbine	4- 6	0,7- 1,1
Condensate pump	2- 7	0,4- 1,1
Feedwater pump	15- 20	0,6- 1,7

Doses of the irradiation on the personnel of VK-50 are within the permissible limits.

3.3 Purification of gas releases.

On the VK-50 the very effective scheme of a burning of the hydrogen and a purification of gas releases are developed and successfully tested during operation [14].

3.4 Passive safety systems

Modern trends in the development of the high safety level of the NPPs are embraced by the simplification of the control systems, use of the natural forces for initiation of the protection actions and by use of the passive systems for an accident control. All such ideas are implemented in the projects of the BWR with NC. Practically new plants meet all safety criteria with passive safety systems- passive inventory control, containment heat removal, fission product control and others.

3.5 In-vessel retention of the BWRs core in severe accident.

Large water supply with the passive actuation in the case of emergency provides adequate cooling of the fuel and ensures the core to be retained in the vessel. Important features of the BWR design in Russia are locations of the control rod drives on the cover of the reactor. This means that on the low head of the reactor there are no connections, no nozzles, no chances for leakage. It could help to retentain the core melt in the reactor vessel in the case of severe accident.

4. Conclusion

BWRs with NC are rather good choice among the various alternatives to get acceptable safety level of NPPs together with the moderate construction and operation expenses. Such reactors could be considered as suitable alternative for reconstruction of the NPPs after the decommissioning of PWRs.

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