ENERGETIKA. 2010. T. 56. Nr. 1. P. 40–47

Removal of decay heat from shut-down RBMK-1500 reactor by natural circulation of water

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The main problem in nuclear energy is providing safety at all stages of the lifetime of nuclear installations in conditions of normal operation, accidents and at shutdown. In the RBMK-1500 reactor, the fuel assemblies remain for a long time inside the reactor core after its final shutdown. The decay heat from reactor the core is removed mainly by natural circulation of water in the reactor cooling system. This paper discusses the reliability of this reactor cooling mode in different accident conditions.

Key words: RBMK-1500 reactor, decay heat, natural circulation

Abbreviations: DS – drum separator, FC – fuel channel, MCP – main circulation pump, NPP – nuclear power plant, RBMK – acronym for a Russian-type graphite moderated boiling water reactor, RCS – reactor cooling system, SWP – steam water piping

1. INTRODUCTION

The Ignalina NPP was the only nuclear power plant in Lithuania. It consisted of two units commissioned in 1983 and 1987. Both units were equipped with channel-type graphite-moderated boiling water reactors RBMK-1500 with the nominal electrical power 1500 MW. Unit 1 of the Ignalina NPP was shut down for decommissioning at the end of 2004, and Unit 2 was operating until the end of 2009.

A detailed description of the Ignalina NPP with RBMK-1500 reactors is presented in a book [1]. The design of the RBMK as a channel-type reactor allows changing the fuel assemblies on-line. This online refuelling puts specifics on the accidents during refuelling, but the integral reactor core characteristic remains almost constant during the reactor operation. Thus, in RBMK reactors, the integral reactivity dependence on fuel burn-up is minimized. Therefore, in RBMK-1500 reactors fuel assemblies remain for a long time inside the reactor core after the final shutdown. Only after a few years the fuel (part of which is relatively fresh) can be removed into spent fuel storage pools.

The pressure in drum separators of the reactor cooling system is atmospheric at reactor shutdown. Even in the shutdown conditions decay heat is generated in the reactor core, and the heat should be removed to keep the temperature below the water saturation temperature (according to the technological regulation of the RBMK-1500, a cooled reactor is considered as a subcritical reactor when the water temperature is not exceed 80 °C in the RCS and the graphite stack temperature is not above 100 °C). Normally, at shutdown the RBMK reactor RCS is filled with water up to connection of steam water piping to the DS. In this case, heat from fuel channels is removed by natural circulation of water (Fig. 1). The hot water from fuel channels (*6*) passes to SWPs (*7*) and DSs (*1*). DSs and SWPs are elevated ~15 m above the reactor core and placed in DSs compartments. The 1661 pipes of the SWP (*7*) with the external diameter 76 mm and approximate length 27.4 m each provide a significant heat transfer area (Fig. 1). From the SWP, heat is removed by air employing the ventilation system of DSs compartments. Such type of reactor cooldown in the technological regulation of RBMK-1500 is called "cooldown in a natural circulation mode"; it allows maintaining the water temperature in the RCS below 100 °C. Because the acceptance criterion for fuel cladding is 700 °C [2], the intactness of fuel cladding is assured.

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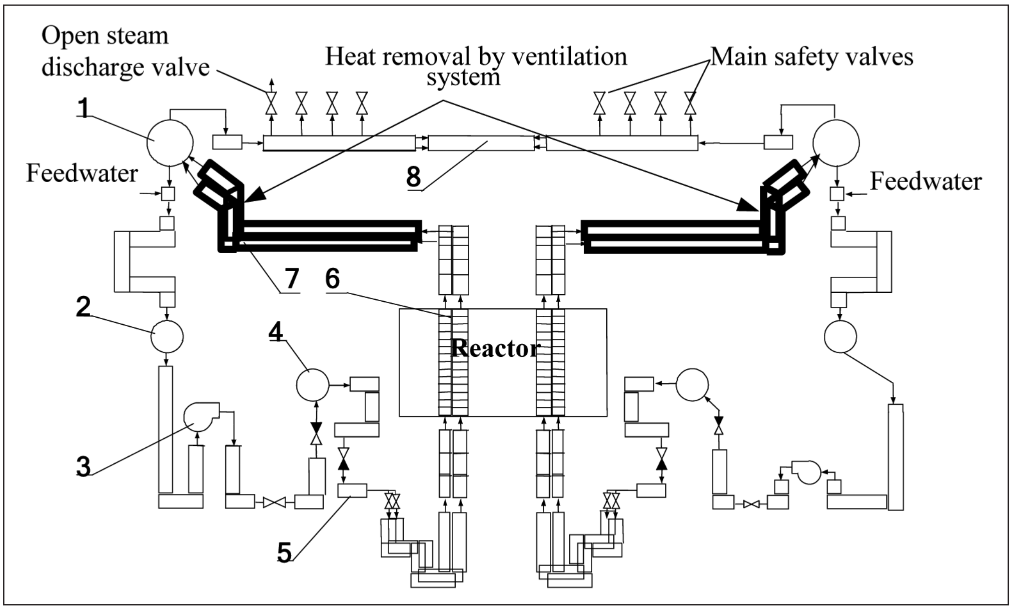

Fig. 1. Schematic representation of RBMK-1500 reactor cooling system during water natural circulation:
1 – DSs, *2* – MCP suction header, *3* – tripped MCPs, *4* – MCP pressure header, *5* – group distribution header, *6* – fuel channel,
7 – steam water piping, *8* – steam lines

The main problem in nuclear energy is providing safety at all stages of the lifetime of nuclear installations in conditions of normal operation, accidents and at shutdown. This paper discuses the reliability of decay heat removal from the RBMK-1500 reactor core after shutdown by natural circulation of water. Three most likely accident conditions leading to a decrease of the coolant flow rate, the loss of cooling system make-up and of heat removal were analyzed using the RELAP5-3D code:

• blockage of water flow rate through fuel channels due to sludge deposits on intensifying grids of fuel assemblies;

• station blackout case;

• inadvertent closure of the steam discharge valve.

2. DEVELOPMENT OF THE RELAP5-3D MODEL FOR THE ANALYSIS OF HEAT REMOVAL FROM THE CORE

The analysis of heat removal from fuel assemblies by natural circulation of water in RCS and transfer to the air ventilation system is performed using the thermal RELAP5-3D hydraulic code [3]. The RELAP5-3D code is an outgrowth of the one-dimensional RELAP5 / MOD3 code developed at the Idaho National Laboratory for the U. S. Nuclear Regulatory Commission. It is a "best estimate" system code suitable for the analysis of all transients and postulated accidents in Light Water Reactor systems, including both large- and small-break loss-of-coolant accidents as well as the full range of operational transients. The code is based on a non-homogeneous and non-equilibrium model for the two-phase system, which is solved by a fast, partially implicit numerical scheme to permit the economic calculation of the system transients. The code includes many generic models allowing to simulate general thermo hydraulic systems. The models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control system logic elements. The RELAP5 code (which is the basis for the RELAP5-3D code) was successfully applied to PWR and BWR reactors. Since 1993, the RELAP5 model of the Ignalina NPP was used in the Lithuanian Energy Institute for the analysis of thermal-hydraulic response of the plant to various transients. During the code adaptation and validation process [4–7], it was shown that not only common phenomena for all types of reactors (heat transfer in fuel assemblies during a post-critical heat flux; counter-current flow; behaviour of circulation pumps in one- and two-phase conditions; mixture level and steam separation in big volumes; coolant blowdown through break, etc.), but also the specific phenomena for channel-type RBMK reactors (flow instability in parallel steam generating channels, radiation heat transfer in reactor channels, natural circulation development and degradation, water entrainment from DS to steamlines, etc.) are modelled adequately in the developed RBMK-1500 model.

In the RELAP5-3D model of the Ignalina plant, both loops of the RCS are represented. Flow paths within a loop are modeled by one or more passes. In turn, a core pass model uses one or more equivalent fuel channels. The equivalent FC allows modelling the heat generation in a group of real channels as well as hydraulic properties of this group. Heat structures of the equivalent FC are modeled by multiple axial and radial control volumes. Steam paths that remove vapor from DSs are represented explicitly, including steam lines, steam discharge valves, etc. The feed water system is represented

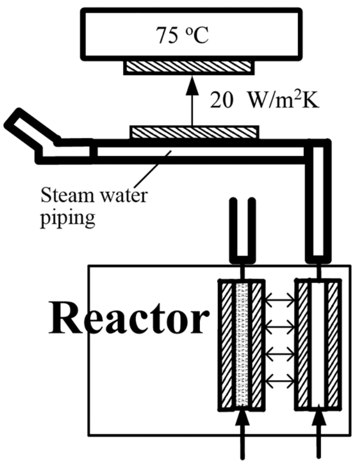

Fig. 2. Simplified nodalization scheme for modelling of heat removal from SWP

explicitly. A more detailed description of the RCS model developed using the RELAP5-3D code is presented in [8, 9].

At modelling it was accepted, that the reactor cooling system is filled with water up to the connection of the steam water piping to the DS. The main circulation pumps are tripped. One steam discharge valve is opened to maintain the atmospheric pressure in the RCS. The temperature of water in the DS is 98 °C, in fuel channels 110–115 °C, of the graphite stack 120 °C. Such conditions are achieved after reactor shutdown, at the end of the reactor cooldown process. According to the RBMK-1500 technological regulation, the reactor could be transferred into the coolant natural circulation mode no earlier than one day after reactor shutdown. It was assumed that the reactor core is loaded with uranium–erbium fuel of 2.6% ^{235}U enrichment at an average burnup depth 25 MW days/kgU (the averaged parameters for the reactor core). The decay heat of the fuel assembly in one fuel channel in such conditions (one day after reactor shutdown) is already reduced from approximately 2500 kW (at normal reactor operation) down to 13.99 kW. The heat level was assumed constant in calculations as a conservative assumption. The reactor power was conservatively accepted as a total decay heat in 1661 averagely loaded FCs. The decay heat from the reactor is removed due to natural circulation of water. The hot water from fuel channels passes to SWPs and DSs. Air ventilation in DS compartments removes heat from the pipes. Water in the RCS does not boil, and no additional make-up by water is required.

The "multidimensional heat conduction" model is employed for modelling heat removal from the external area of the steam water piping. The main idea of the modelling is presented in Fig. 2. Steel walls of the steam water piping are modelled by heat structures. The air volume in DS compartments is modelled by air volume with a constant temperature of 75 °C. The heat structure of this volume is coupled with the heat structure of the SWP through an "air gap", using the multidimensional RELAP5-3D heat conduction model. The heat transfer coefficient of this gap is $\alpha = 20\ \mathrm{W/m^2 \cdot K}$ (this assumption represents the forced air circulation in the compartment due to the operation of the ventilation system).

3. ANALYSIS OF WATER FLOW BLOCKAGE THROUGH FUEL CHANNELS

Blockage of several fuel channels due to sludge deposits on intensifying grids of fuel assemblies is the most likely event initiating a decrease of the coolant flow rate through fuel channels. Sludge deposits lead to a reduction of the open flow area in a fuel channel, as well as can create a zone of local stagnation of water flow in its separate parts. As there is no possibility to define the possible local effects, conservatively we accept that due to sludge deposits on intensifying grids of fuel assemblies, the open flow area of FCs decreases by half. It is accepted that the open flow area of FCs decreases in all fuel channels on the whole height of these channels.

While modelling, in the beginning it is considered how such a reduction (by half) of the open flow area in an FC would affect the cooling of fuel assemblies under conditions of a normal operation of the reactor (reactor power is 4200 MW, three MCPs operate in each loop of the RCS). The analysis is performed for a single FC with a power of

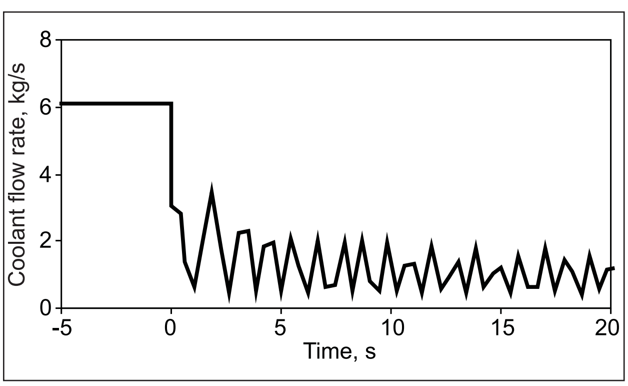

Fig. 3. Sharp decrease (by half) of the open flow area in FC (normal operation of the reactor, three operating MCPs in one loop of the RCS). Behaviour of coolant flow through a single fuel channel with the power of 3.28 MW

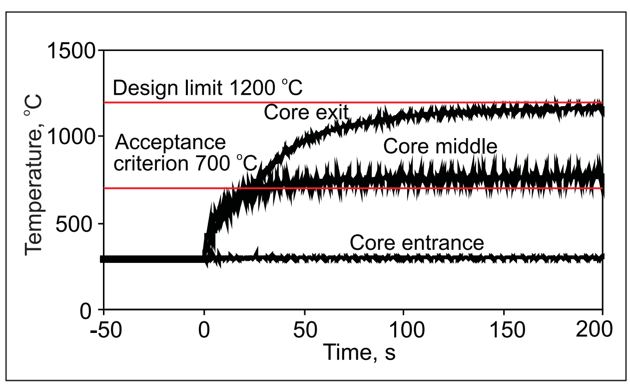

Fig. 4. Sharp decrease (by half) of the open flow area in FC (normal operation of the reactor, three operating MCPs in one loop of the RCS). Behaviour of fuel cladding temperature in a fuel channel with the power of 3.28 MW

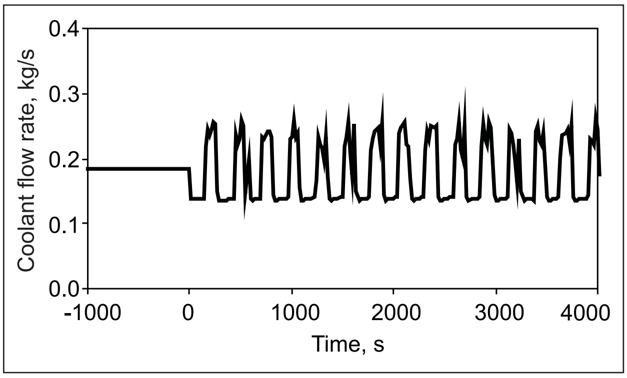

Fig. 5. Sharp decrease (by half) of the open flow area in FC at shutdown and in cooled reactor. Water flow behaviour in a single fuel channel

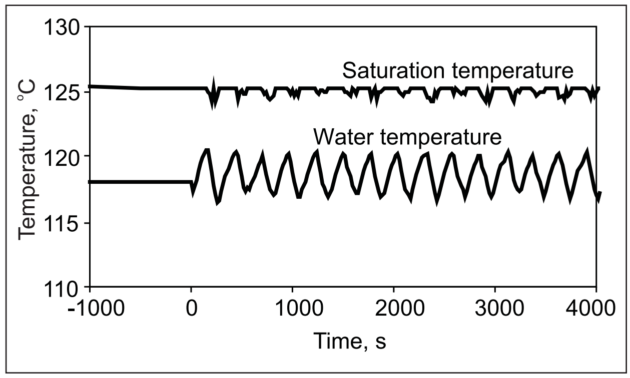

Fig. 6. Sharp decrease (by half) of the open flow area in FC at shutdown and in cooled reactor. Behaviour of water temperature in fuel channel

3.28 MW. In Fig. 3, the coolant flow rate through an FC under a sharp reduction (by half) of the open flow area in an FC (from $2.273 \cdot 10^{-3}$ mm^2 to $1.1365 \cdot 10^{-3}$ mm^2) is presented. As is seen in the figure, after reduction of the open flow area in an FC, the coolant flow rate through the channel sharply decreases and becomes unstable. The insufficient heat removal by the coolant initiates a boiling crisis in the FC. The temperature of fuel cladding starts increasing (Fig. 4), and as soon as after about 30 seconds the maximal temperature of the fuel cladding reaches the acceptance criterion of 700 °C [2]. In a long-term, the maximal temperature of the fuel cladding comes closer to the design limit of 1200 °C [2]. Thus, results of the analysis have shown that reduction (by half) of the open flow area in FCs damages (by overheating) the fuel claddings.

In Figs. 5 and 6, the situation for the open flow area reduction at shutdown and in the cooled reactor is presented. The initial condition of the RCS has been discussed above (it is filled with water, the MCPs are tripped, the temperature of water in the DS reaches 98 °C, the pressure in the DS is atmospheric, the temperature of water in FCs is 110–115 °C, of the graphite stack 120 °C, the decay heat from the reactor is removed by ventilating the DS compartments). As one can see in Fig. 5, while modelling, the accepted sharp reduction (by half) of the open flow area in the FC leads to an unstable natural circulation of water through the channel. The instability of the water flow rate causes fluctuations of water temperature in the FC; however, the temperature of water remains below the boiling point (Fig. 6).

Thus, results of the analysis have shown that reduction (by half) of the open flow area in FCs (due to sludge deposits on intensifying grids of fuel assemblies) at shutdown and at the cooled reactor leads to a insignificant, although acceptable, deterioration of fuel cooling conditions.

4. ANALYSIS OF THE POWER PLANT BLACKOUT

For the analysis of the consequences of a beyond-design accident – the power plant blackout during reactor cooling in a natural circulation mode – it was assumed that at the moment of time t = 0 seconds the operation of the ventilation system in DSs compartments stops. The make-up of the reactor cooling system by the water-using design means is unavailable due to the total loss of electric power supply (station blackout). At the initial stage, the temperature of water in FC is 110–115 °C, and the temperature of the graphite stack is 120 °C (Fig. 7). After switching off the ventilation system,

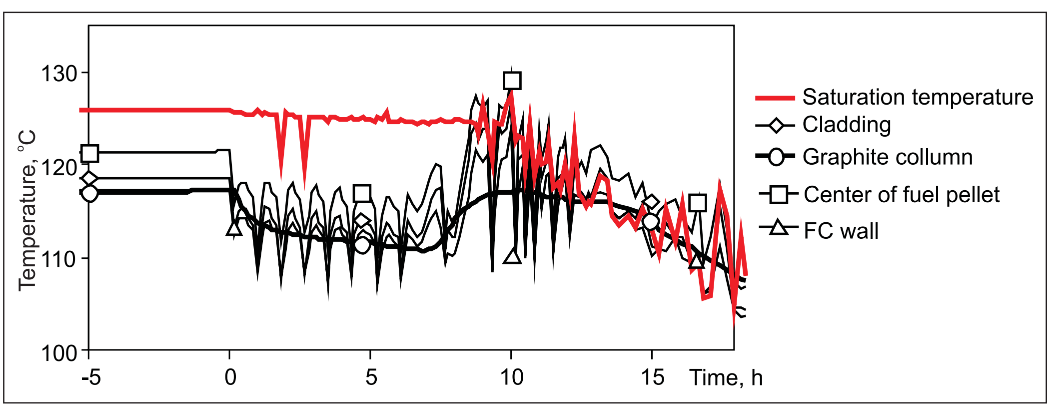

Fig. 7. Plant blackout during reactor cooling by coolant natural circulation mode.
Behaviour of temperature of the core components

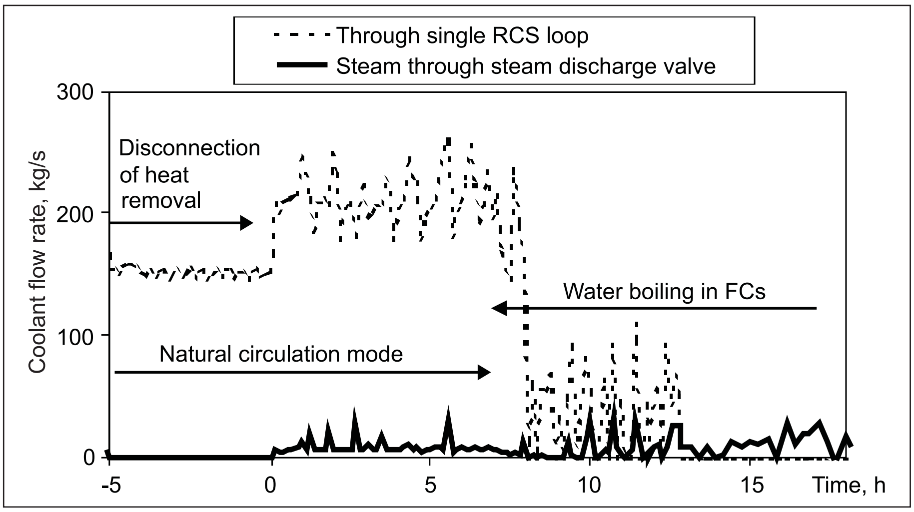

Fig. 8. Plant blackout during reactor cooling by coolant natural circulation mode. Behaviour of coolant flow through FCs of one RCS loop

water in the SWP will not be cooled. Water, its temperature being 115 °C, passes into DSs where the pressure is atmospheric. Water in DSs boils, and the steam is discharged through the opened steam discharge valve (this valve is opened at the shut-down reactor to maintain atmospheric pressure in the RCS). After switching off the ventilation system, the water flow rate through FCs slightly increases (Fig. 8) because the heat removal rises upwards (from the SWP passes into the DS) and thus increases the driving force. As the steam is discharged into steam condensing pools and there is no additional make-up of RCS by water, the volume of water in the DS decreases (Fig. 9). DSs are emptied in 8 hours after the loss of power supply (Fig. 9); simultaneously, the coolant natural circulation is terminated (Fig. 9), because after emptying DSs the circuit of circulation interrupts. After termination of the coolant natural circulation, the boiling of water in FCs begins, as is seen in Fig. 7 (the temperatures of the core components match the temperature of water saturation).

After the beginning of water boiling in FCs (8 hours after the loss of power supply), the pressure of the water column in the FC–DS path starts to decrease because part of water has evaporated. This leads to a decrease of pressure in fuel channels (Fig. 10). After the water boiling beginns in FCs, the heat transfer coefficient from the fuel assembly to the coolant somewhat decreases (Fig. 11), but remains sufficient for a reliable cooling of fuel assemblies (the temperature of the core components remains within the limits of 130–100 °C) (Fig. 7). At water boiling in FC, the temperature of the core components even decreases because of pressure drop in FCs (Fig. 10) and reduction of the water saturation temperature (Fig. 7).

Thus, in case of blackout during the reactor cooling by coolant natural circulation mode, the dryout of fuel assemblies can occur not earlier than 18 hours after the beginning of the accident. Before this, the fuel is reliably cooled by coolant natural circulation mode and water boiling in fuel channels with steam removal through the steam discharge valve. The operators have time enough to find the possibilities to provide a make-up of the RCS by water from non-regular, non-designed water sources.

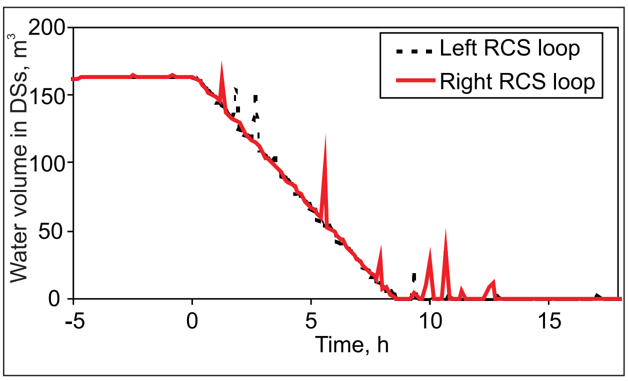

Fig. 9. Plant blackout during reactor cooling by coolant natural circulation mode. Water volume behaviour in both DSs of one RCS loop

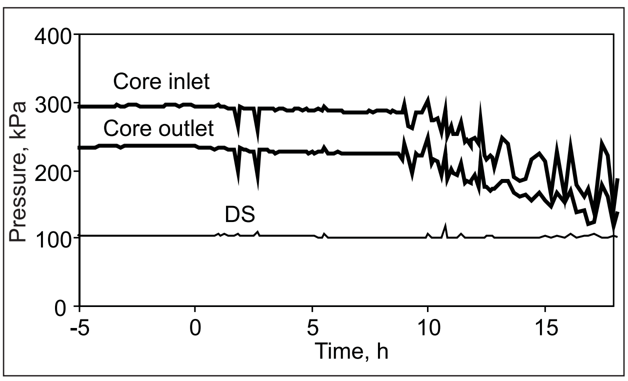

Fig. 10. Plant blackout during reactor cooling by coolant natural circulation mode. Behaviour of pressure in RCS

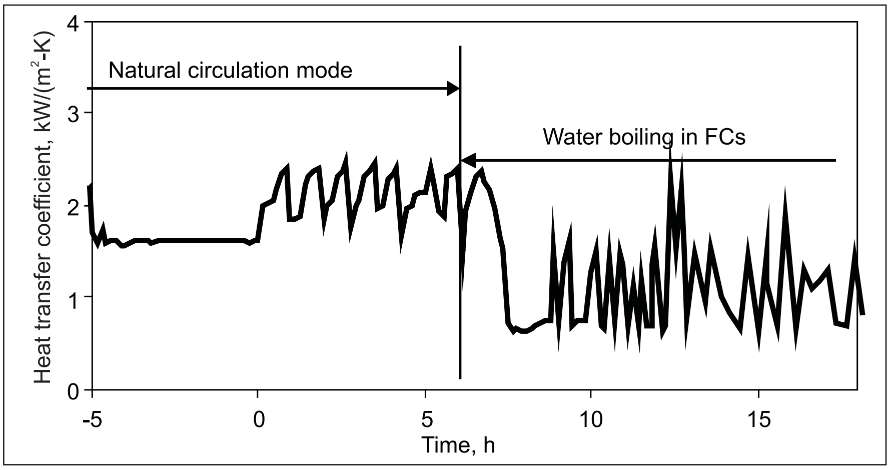

Fig. 11. Plant blackout during reactor cooling by coolant natural circulation mode. Change of the coefficient of heat transfer from fuel assembly to the coolant

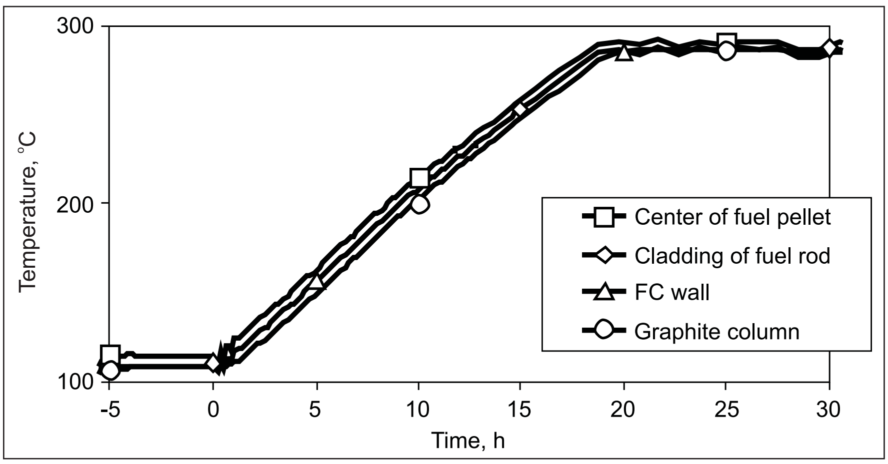

Fig. 12. Inadvertent closure of steam discharge valve and trip of ventilation system in DS compartments during reactor cooling by coolant natural circulation mode. Behaviour of temperature in the core components

5. ANALYSIS OF INADVERTENT CLOSURE OF STEAM DISCHARGE VALVE

During the modelling, it was assumed that at the time moment t = 0 seconds the steam discharge valve, which maintains atmospheric pressure in the RCS, is closed. Also, the operation of the ventilation system in DS compartments stops simultaneously. Thus, all sources of heat removal from fuel assemblies are eliminated. After the termination of heat removal, all decay heat accumulates in RCS water and in core components. The temperature of these components and, thereby, the coolant starts to increase (Fig. 12) and water boils. Because of the generated steam, the pressure in RCS starts to increase (Fig. 13), because in this case the circulation circuit is a closed system.

With the boiling of water, because of the resulting steam–water mixture, the water level in DSs starts to increase

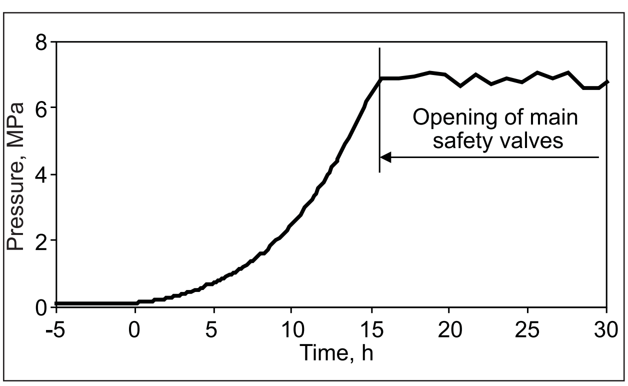

Fig. 13. Inadvertent closure of steam discharge valve and trip of ventilation system in DS compartments during reactor cooling by coolant natural circulation mode. Behaviour of pressure in DSs

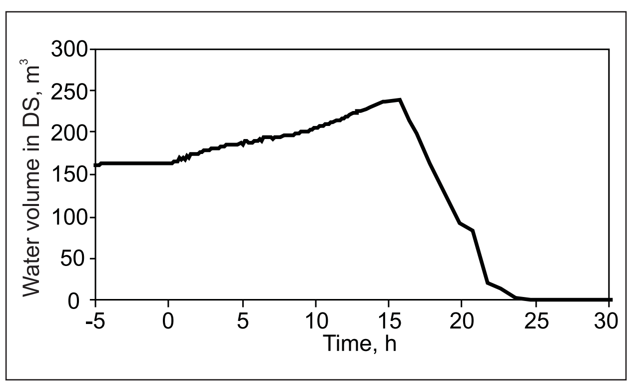

Fig. 14. Inadvertent closure of steam discharge valve and trip of ventilation system in DS compartments during reactor cooling by coolant natural circulation mode. Behaviour of water volume in both DSs of one RCS loop

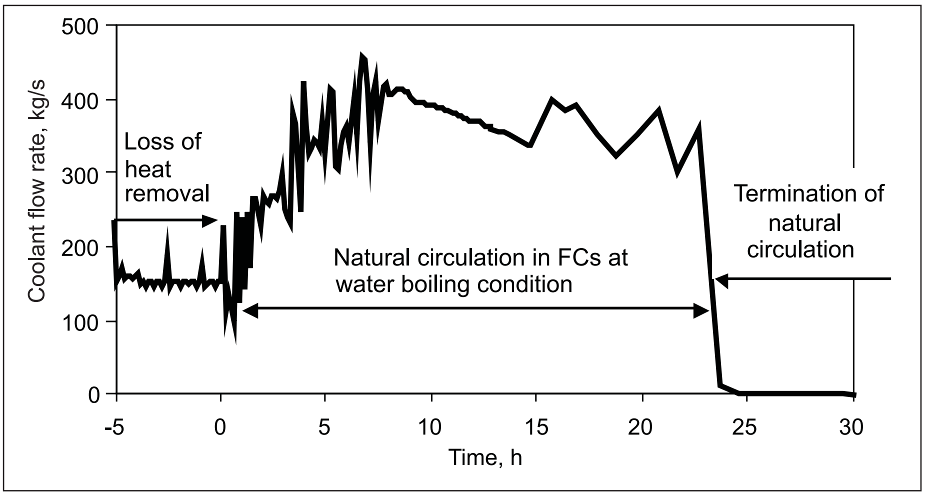

Fig. 15. Inadvertent closure of steam discharge valve and trip of ventilation system in DS compartments during reactor cooling by coolant natural circulation mode. Behaviour of coolant flow rate through one RCS loop

(swell – see Fig. 14). Upon achieving the set-point for operation of the main safety valves, the first group of these valves starts to open. Steam discharge through safety valves maintains an approximately constant pressure in DSs (Fig. 13). The steam is discharged through safety valves into steam-condensing pools; therefore, the volume of water in DSs after the beginning of operation of the main safety valves starts to decrease (Fig. 14).

The behaviour of the coolant flow through the fuel channels of one RCS loop is presented in Fig. 15. One can see that at the operating ventilation system the total water flow rate in the coolant natural circulation is approximately 150 kg/s through one RCS loop. After closing the steam discharge valve, because of the increasing coolant temperature, the flow rate of the coolant natural circulation increases also. Termination of natural circulation occurs after emptying DSs, which leads to a break of the circulation loop.

Results of the analysis have shown that in case of terminating heat removal from the RCS by switching off the ventilation in DS compartments as well as closing the steam discharge valve during reactor cooling by the natural circulation mode, the pressure in the RCS increases up to achieving set-points for the main safety valve operation. However, the temperature of the core components does not exceed 300 °C. Such cooling of fuel assemblies will proceed up to water evaporation from DSs. If there is no possibility to restore heat removal from the RCS, the operator should maintain the water level in DSs and provide an additional make-up of the RCS by water. For this purpose, there is a sufficient stock of time: emptying of DSs starts approximately 15 hours after the trip of ventilation and closure of the steam discharge valve. The decay heat from fuel assemblies will be removed by discharging steam through the main safety valves. If the make-up of the RCS is not restored, the drying out of fuel assemblies can occur 15–18 hours later after the emptying of DSs.

6. CONCLUSIONS

The analysis performed using the RELAP5-3D code shows that the RBMK-1500 reactor after shutdown can be cooled by natural circulation of water. Three most likely accident conditions leading to a decrease of the coolant flow rate, loss of the cooling system make-up and loss of heat removal were analyzed:

- blockage of water flow through fuel channels due to sludge deposits on intensifying grids of fuel assemblies;
- the plant blackout case;
- inadvertent closure of the steam discharge valve.

The results of the analyses have demonstrated that a decrease of coolant flow rate through fuel channels is very unlikely: even in the case of open flow area reduction by half, the fuel assemblies are reliably cooled, and water temperature inside the fuel channels remains below saturation. In the loss of cooling system make-up and of heat removal (plant blackout and inadvertent closure of the steam discharge valve), the operators have time enough to find the possibilities to provide the make-up of the RCS by water, using non-regular, non-designed water sources. In the case of inadvertent closure of the steam discharge valve and loss of ventilation in DS compartments, the emptying of DSs starts approximately 15 hours after the beginning of the accident. The dryout of fuel assemblies can occur 15–18 hours after DS emptying, if the make-up of RCS is not restored. Thus, the decay heat from the RBMK reactor after shutdown can be reliably removed by natural circulation of water in the reactor cooling system.

Received 18 August 2009
Accepted 23 December 2009

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LIEKAMOSIOS BRANDUOLINIO SKILIMO ŠILUMOS ŠALINIMAS SUSTABDYTAME RBMK-1500 REAKTORIUJE NATŪRALIOS VANDENS CIRKULIACIJOS BŪDU

S a n t r a u k a

Pagrindinė branduolinės energetikos problema yra tai, kad sauga turi būti užtikrinta visą branduolinės įrangos gyvavimo laiką: eksploatacijos metu, avarijų atvejais ar ją sustabdžius. RBMK-1500 reaktoriuje kuro rinklės po galutinio jo sustabdymo ilgą laiką bus laikomos reaktoriaus aktyviojoje zonoje. Liekamoji šiluma nuo reaktoriaus aktyviosios zonos šalinama daugiausia natūralios vandens cirkuliacijos reaktoriaus aušinimo sistemoje būdu. Straipsnyje aptariamas šio reaktoriaus aušinimo būdo patikimumas įvairiais avariniais atvejais.

Raktažodžiai: RBMK-1500 reaktorius, liekamoji šiluma, natūrali cirkuliacija

Альгирдас Калятка, Эугениюс Ушпурас, Миндаугас Вайшнорас, Георгий Кривошеин

СНЯТИЕ ОСТАТОЧНОГО ТЕПЛА ЯДЕРНОГО РАСПАДА ОТ ЗАГЛУШЕННОГО РЕАКТОРА РБМК-1500 С ИСПОЛЬЗОВАНИЕМ ЕСТЕСТВЕННОЙ ЦИРКУЛЯЦИИ ВОДЫ

Р е з ю м е

Основной проблемой в ядерной энергетике является обеспечение безопасности в течение всего времени жизни ядерной установки: во время эксплуатации, при аварийных условиях и после остановки. В реакторах РБМК-1500 тепловыделяющие сборки после окончательного останова длительное время остаются в активной зоне. Остаточное тепловыделение от активной зоны реактора в основном отводится с помощью естественной циркуляции воды в системе охлаждения реактора. В данной статье обсуждается надежность этого способа охлаждения реактора при разных аварийных условиях.

Ключевые слова: реактор РБМК-1500, остаточное тепловыделение, естественная циркуляция