Accident management for RBMK-1500 in the case of loss of long-term core cooling

Algirdas Kaliatka

Laboratory of Nuclear Installation Safety, Lithuanian Energy Institute, Breslaujos 3, LT-3035, Kaunas, Lithuania

Results of the Level 1 Probabilistic Safety Assessment (PSA) of the Ignalina Nuclear Power Plant (NPP) have shown that in the topography of risk, transients dominate above accidents with the loss of the coolant accidents. PSA has shown that failure of the core long-term cooling is the main contributor to the frequency of the core damage. However, transition to the condition of the reactor core due to a loss of the long-term cooling specifies potential opportunities for the management of the accident consequences.

This paper presents a detail thermal-hydraulic analysis of the long-term core cooling accidents, performed using the RELAP5 model of Ignalina NPP reactor primary circuit and plant safety systems. On the basis of this analysis the accident management strategy was developed.

Key words: RBMK-1500, long-term core cooling accidents, accident management strategy

1. INTRODUCTION

Results of PSA of the Ignalina NPP have shown that failure of the core long-term cooling is the main contributor to the frequency of the core damage. On the other hand, there is a sufficient time for employing different measures to mitigate the consequences of an accident. Hence, accident management for the mitigation of the accident consequences should be considered and developed.

The most likely initiating event, which probably leads to the loss of long-term cooling, is station blackout. The station blackout is the loss of preferred electrical power supply for local needs with an additional failure on start-up of all diesel generators. In the case of loss of electrical power supply the Main Circulation Pumps (MCP), service water pumps and pumps for feedwater supply are switched-off. Failure of diesel generators leads to non-operability of all these pumps during a long period of time (hours). It means that it is impossible to feed the reactor primary circuit by water.

Analysis of station blackout for the Ignalina NPP was performed using the RELAP5 code. The results showed that approximately one and a half hour after beginning of the accident begins a dangerous heating-up of fuel elements and Fuel Channel (FC) tubes.

Three ways of potential accident management for loss of the long-term core cooling are discussed in this paper:

- decay heat removal from the core by ventilation of Drum Separator (DS) compartments
- decay heat removal from the core by direct water supply into the Reactor Cavity (RC)
- de-pressurisation of the reactor coolant system and water supply using non-regular means.

2. IGNALINA NPP RELAP5 MODEL

The Ignalina NPP is a twin-unit with two RBMK-1500, graphite moderated boiling water multichannel reactors. Several important design features of RBMK-1500 are unique and extremely complex in respect to western reactors: the fuel clusters are loaded into individual channels rather than a single pressure vessel, the plant can be refuelled on-line, the neutron spectrum is thermalized by a massive graphite moderator block. The RBMK-1500 coolant loop, having a very long flow length of more than 200 m, consists of 1661 of parallel FC and numerous components, such as headers, pumps, valves, etc. A brief description of the Main Circulation Circuit (MCC) and plant safety systems of the Ignalina NPP is given in [1].

Analysis of the accidents in the RBMK-1500 is performed using the best-estimate code RELAP5. The original RELAP5 computer code has been de-

ISSN 0235–7208. E n e r g e t i k a . 2003. Nr. 1

veloped by Idaho National Engineering Laboratory. This is a one-dimensional non-equilibrium two-phase thermal-hydraulic system code. RELAP5 code has been successfully applied to pressurised water reactors. At the Lithuanian Energy Institute this code has been used since 1992, and it has been adapted to simulate the RBMK type reactors. The key features of the Ignalina plant RELAP5 model used for the loss of long-term core cooling analysis are as follows:

- The MCC is represented by a group of equivalent fuel channels. The equivalent fuel channel models heat generation in a group of real channels, as well as hydraulic properties of this group. The equivalent fuel channels are modelled by multiple axial and radial control volumes.
- The circuit of the Control and Protection System (CPS) rods cooling and radial reflector cooling is modelled explicitly.
- Heat transfer among the equivalent fuel channels and CPS channels is approximated by means of heat exchange through the graphite moderator gaps to the RC gas.
- The heat structures for modelling heat sink from equipment located in DS compartments and from surfaces of pipelines are added to models of drum separators, steam-water communications and steamlines.
- Steam paths that remove the vapour from drum separators are represented explicitly, including steam lines, steam discharge valves, etc.
- RC formed on a metal structure of the reactor shell together with bottom and top metal plates is modelled.

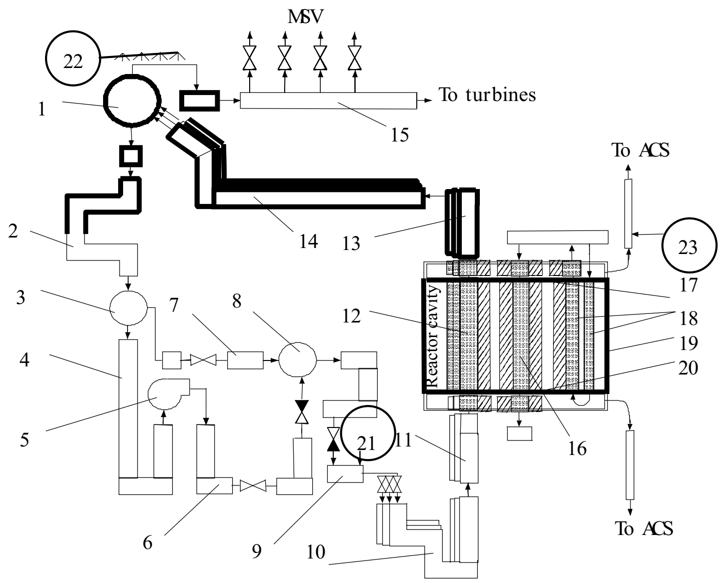

Fig. 1. RELAP5 Ignalina NPP model nodalization scheme: *1* – DS, *2* – downcomers, *3* – MCP suction header, *4* – MCP suction piping, *5* – MCPs, *6* – MCP discharge piping, *7* – bypass line, 8 – MCP pressure header, *9* – GDHs, *10* – lower water communication line, *11* – reactor core inlet piping, *12* – FC, *13* – reactor core outlet piping, *14* – steam–water communication line, *15* – steam line, *16* – CPS rod cooling channel, *17* – top metal plate, *18* – radial reflector cooling channel, *19* – reactor shell, *20* – bottom metal plate, *21* – water supply to GDH, *22* – water supply to spray system for air humidification in DS compartments, *23* – water supply directly to the reactor cavity

The nodalization scheme of the Ignalina NPP RELAP5 model is presented in Fig. 1. The MCC is simulated by one loop. Such simplification is possible due to the fact that both loops in the MCC in the case of loss of long-term cooling are actually in the same conditions. This one core pass is represented by three equivalent fuel channels (maximal, average and minimal power). For the thermal core power of 4200 MW, the channel average power is assumed to be 2.53 MW, the maximum channel power is 3.75 MW, and the minimum channel power is 0.88 MW.

Heat structures of the equivalent fuel channel simulate the active region in the reactor core. The fuel element is modelled with an equivalent four radial node model. One of these radial nodes is for the fuel pellet, one for the gap region and two for the cladding. The vertical bundle option is used in heat structure description of fuel assembly with 18 fuel elements. The fuel channel and the graphite stack are modelled with an equivalent six radial node model. Two of these radial nodes are for the fuel channel wall, one for the gap and graphite rings region and three for the graphite blocks. Fuel element, fuel channel, graphite rings and graphite blocks are modelled with 14 axial segments, 0.5 m long each. The square graphite stack is represented by an equivalent cylindrical volume. Approximately 95% of the total energy is deposited in the fuel and 5% in the graphite.

Fuel channels are connected on one end to the Group Distribution Header (GDH) (9) by the lower water communication lines (10) via the water flow control valve. The other end of the equivalent channels is connected to the DS by the steam–water communication line (14). Four real DSs are modelled as one volume (1). All downcomers are represented by a single equivalent pipe (2), further subdivided into a number of control volumes. MCP suction (3) and pressure (8) headers are represented as branch objects. Six operating MCPs are represented by one equivalent element (5) with check and throttling-regulating valves. Stand-by MCPs are not modelled. The bypass line (7) between the MCP suction and pressure headers is modelled with the manual valves closed. This is in agreement with a modification recently implemented at the Ignalina NPP. Steam from the separators is directed to the turbines via steam lines (15). Two "servo valves" [2] simulated steam supply to the turbines. The control of these valves was modelled on the algorithm of steam pressure regulators used at the Ignalina NPP. Steam pressure is controlled and peaks of pressure are eliminated by Main Safety Valves (MSV). These valves are modelled by "motor valve" [2] elements with a corresponding algorithm of theirs opening and closure.

All 235 CPS rods cooling channels are modelled by one equivalent channel (16). The elements (18) simulate 156 radial reflector cooling channels having a "field's pipe" design. Water into these channels is supplied from the top distribution header and removed into the bottom distribution header. The reactor cavity is formed by a metal structure of the reactor shell (19) together with the bottom (20) and top (17) metal plates. The fuel channels and CPS channels are allocated inside the holes of graphite columns. There are 2488 graphite columns, which construct the reactor graphite stack. The graphite stack is simulated as a thermal structure in the presented RELAP5 model. The circuit of the CPS rods (16) cooling and radial reflector (18) cooling is simulated explicitly. This approach considers removal of decay heat from the graphite stack to the CPS cooling circuit.

The developed model considers three different ways of water supply:

- water supply from Emergency Core Cooling System (ECCS) hydro-accumulators, deaerators or using non-regular means to supply water to the GDH (21),
- water supply from non-regular means to spray system (22) for air humidification in the DS compartments,
- water supply from non-regular means directly to the reactor cavity (23).

In the RELAP5 model, the water packing scheme normally is used only for specific components (*e.g.*, "pressuriser"). Therefore in all volume control flags, the water packing is turned off for all elements. In the model for all MCC pipelines, the vertical stratification model is turned on. In the elements that model CPS and ECCS pipelines, the vertical stratification is turned off. The wall friction model is turned off in the elements with a large water volume (elements modelling DS and headers) where wall friction has actually no influence. The wall friction is turned on in all other elements. The counter current flow model was used in the fuel channels and elements that simulated the reactor cavity. The choking flow model is applied for the elements of steam discharge valves.

In order to provide confidence in the ability of the models to represent correctly the plant response to the stress conditions, the models have been benchmarked for several operational events, such as trip of all MCPs and spurious opening of three MSVs, inadvertent actuation of ECCS, etc. Calculation results obtained using the Ignalina NPP RELAP5 model agree well with the plant data when similar boundary conditions are imposed [3–5].

3. ANALYSIS OF PLANT BLACKOUT WITHOUT OPERATOR INTERVENTION

The Ignalina NPP belongs to the category of "boiling water" reactors. During normal reactor operation the coolant is supplied by MCP into the fuel channels. In the FC the cooling water is brought to boiling and is partially evaporated. The steam–water mixture then proceeds to the DS, the elevation of which is above the reactor. Here the water settles, while the steam proceeds to the turbines. The feed water from the deaerators is supplied to the MCC by Main Feed Water Pumps (MFWP). In the case of loss of electrical power supply MCPs, the circulating pumps of the service water system and MFWPs are switched off. At the same time, Turbine Control Valves (TCV) of both turbines are closed within 0.4 seconds. The reactor is shut down immediately according to the signals of TCV closure, MCP trip or loss of feedwater supply. The closure of TCV leads to the imbalance between steam removal and steam generation in the core. The pressure in the DS starts to increase. The excessive steam is discharged through the safety valves, thus MCC pressure is maintained at a proper level (Fig. 2). During the first seconds the coolant is supplied by MCP, which have a massive flywheel in order to prolong rotation. Later on, decay heat from the reactor core is removed by a natural circulation of the coolant.

The emergency power supply for the Ignalina NPP is provided by six diesel generators per unit. In the event of a loss of normal electrical power supply they are started up automatically following the trip of all turbines or loss of grid and can supply emergency loads in about 35 seconds. The diesel generators provide power for the emergency feedwater supply pumps and ECCS pumps. The station blackout is the loss of normal electrical power supply with an additional failure on the start-up of all diesel generators. Failure of the diesel genera-

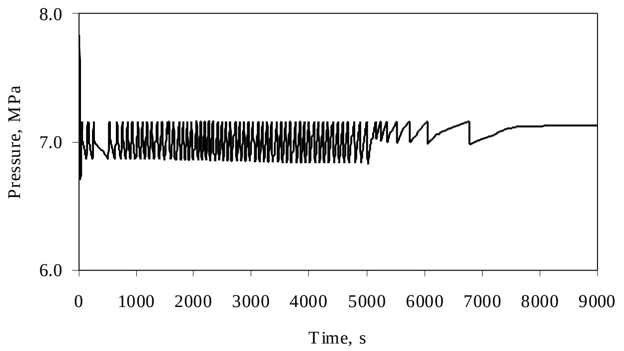

Fig. 2. Station blackout without operator intervention. Pressure in the DS

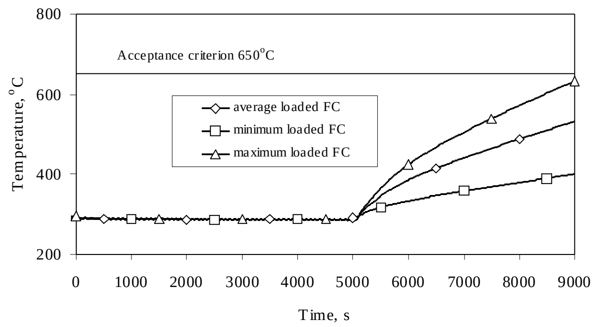

Fig. 4. Station blackout without operator intervention. Peak channel wall temperature in the FC of different power

tors leads to the non-operability of all electrical equipment, except the main safety valves which are connected to batteries. The feeding of MCC by water using regular means is impossible.

The station blackout analysis with RELAP5 showed that in case of no operator intervention, approximately 2000 s (half an hour) after beginning of the accident, DS become empty. One hour later, the dry-out in the core starts. The heat transfer coefficient from fuel rods to the coolant decreases. When the reactor is operated in the steady state conditions (while MCPs are in operation), the heat transfer coefficient is 36–64 kW/m^2·K. After MCP trip and shutdown of the reactor, the heat transfer coefficient decreases rapidly down to 8–16 kW/m^2·K. After core uncovering, the heat transfer coefficient from fuel cladding to the superheated steam equals only 93–110 W/m^2·K. It causes the heating-up of fuel elements and FC tubes. The acceptance criterion for fuel element claddings (700 °C) in the channels with maximum power is reached approximately within 6000 s (more than 1.5 hours) after the beginning of the accident (Fig. 3). Approximately 9000 s (2.5 hours) after the beginning of the accident the safety criterion for FC tube walls (650 °C) is reached (Fig. 4). Since there are 11 maximum loaded fuel channels and the pressure in the MCC is close to the nominal value, the possibility of a rupture of several fuel channels cannot be excluded.

4. ANALYSIS OF DECAY HEAT REMOVAL FROM THE REACTOR CORE BY VENTILATION OF DRUM SEPARATOR COMPARTMENTS

Equipment and piping in the DS compartments have a considerable area of cooling: DS, steam header, part of the steam lines, part of the downcomers, steam–water piping and part of the channels, which are above the core. DS compartments are connected by a corridor of steam–water communication. Both compartments at the top have five rupture panels, 20 m^2 in each DS compartment. Each DS compartment at the bottom has four doorways of 1.2 m^2. DS compartments are connected to the reactor hall through the gaps between the biological shielding blocks. According to the design the effective area of gaps is 5 m^2.

During normal reactor operation the maximal air temperature in the DS compartments is approximately 270 °C. If the mentioned doors and rupture panels are open, natural air circulation will be created in the DS compartments. This natural circulation will be able to remove part of the decay heat ge-

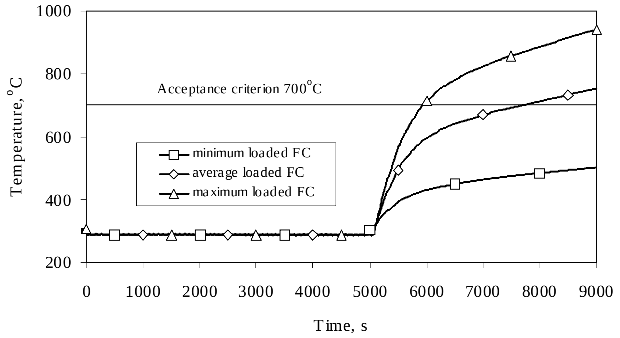

Fig. 3. Station blackout without operator intervention. Peak cladding temperature in the FC of different power

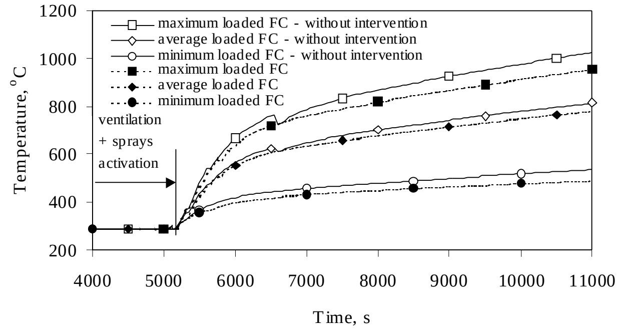

Fig. 5. Analysis of station blackout with ventilation of DS compartments. Peak fuel cladding temperature in the fuel channels of different power

nerated in the core. The results of analysis show that the amount of removed heat is approximately seven times smaller than of heat generated in the core. The decrease of fuel cladding and FC wall temperatures due to ventilation is negligible – the peak fuel cladding temperature after the operator intervention decreases only by 10–30 °C (Fig. 5). Thus, heat removal by means of ventilation is not effective.

5. DECAY HEAT REMOVAL FROM THE RBMK-1500 CORE BY DIRECT WATER SUPPLY TO REACTOR CAVITY

The bulk of residual heat during long-term accidents is generated in the graphite bricks. On the first sight it is reasonable to cool down the graphite. Since the pressure in the reactor cavity is close to the atmospheric, it is possible to use any non-regular sources at low pressure for water supply to cool the graphite bricks. The aim of the performed analysis was to evaluate the expediency of such kind of core cooling in order to assess the possibility to escape the heat-up and melting of the fuel rods, when other measures fail to provide required cooling. This kind of cooling would be used only in the critical case.

By getting in the top part of RC, water flows down along the outer surface of the graphite stack into a ring space between the reactor shell and stack. Gaps between the graphite columns are narrow (1.2 mm), and the temperature of the graphite blocks at the centre of the core is above 300 °C when water supply starts. Thus, water flow downwards along these gaps is practically impossible. Through the area outside of the graphite stack the water flows into the bottom part of the reactor cavity and through the pipelines is removed to the leak-tight compartments of ALS.

The analysis shows that injected water cools the metal structures of RC, bottom and top graphite moderator blocks and the outer surface of the radial graphite reflector. However, because the water does not reach deep lines of the graphite blocks, the temperature of graphite at the centre of the core increases. Changes of peak temperature of the fuel cladding without intervention of the operator are compared with an attempt to cool down the reactor core by direct water supply into RC. As is shown in Fig. 6, the rate of the fuel cladding heat up decreases because of water supply. However, the real reactor cool down does not start. Results of the analysis clearly indicate that the proposed solution concerning heat removal by direct water supply into RC is not feasible.

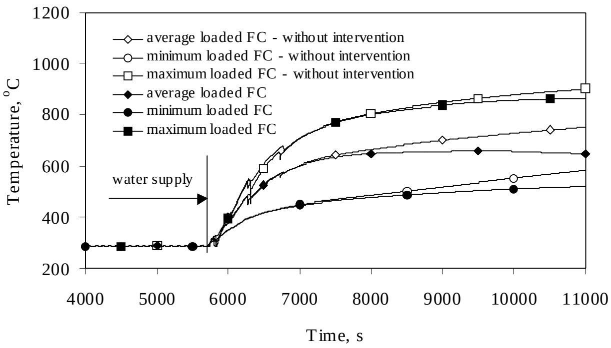

Fig. 6. Analysis of station blackout with direct water supply to reactor cavity. Peak fuel cladding temperature in the fuel channels of different power

6. ACCIDENT MANAGEMENT BY DE-PRESSURISATION OF THE REACTOR COOLANT SYSTEM AND WATER SUPPLY USING NON-REGULAR MEANS

In the case of loss of long-term cooling there are no possibilities for automatic feeding of the reactor with water. But the situation can be seized by the operator's efforts to supply an additional amount of water into the reactor circuit. Additional amount of water allows to stave off the moment of the total water evaporation from the core. In case of loss of preferred electrical power supply, ECCS hydro-accumulators and deaerators are the sources of such additional amount of water. There are about 212 m^3 of water in 16 ECCS hydro-accumulators compressed up to 9 MPa and 480 m^3 of water in four deaerators. The pressure in the deaerators is 1.2 MPa.

In the performed analysis it was assumed that within one hour after beginning of the accident the operator opens the fast-acting valves on the water supply piping from ECCS hydro-accumulators to the GDH (Fig. 7). However, soon the pressure in the accumulators and GDH equalizes and cold water supply is stopped. Approximately 6100 s from the beginning of the accident the fuel cladding temperature in fuel channels starts to increase. In the model it was assumed that the temperature increase is a signal for the operator to initiate de-pressurisation. MCC de-pressurisation is performed opening one MSV. During de-pressurisation of MCC, water supply from ECCS hydro-accumulators is restored.

Water supply from deaerators becomes possible only after pressure in GDH decreases down to 1.2 MPa (*i.e.* down to pressure level in the deaerators). Water from the deaerators reaches the core, heats up there, boils up and thus maintains the MCC pressure. Such a pressure-maintaining process lasts approximately one hour. Further, due to a decrease of water level in the deaerators, water supply to GDH is reduced and the pressure in MCC starts to decrease.

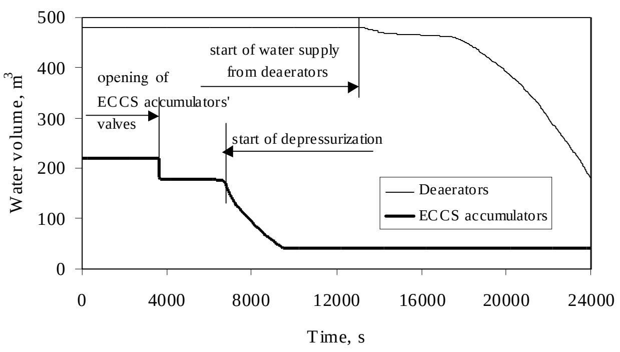

Fig. 7. Analysis of station blackout with de-pressurization of MCC and water supply using non-regular means. Water volume in the ECCS accumulators and deaerators

As is shown in Fig. 8, a repeated increase of fuel cladding temperature starts more than three hours after the beginning of the accident. Water supply from the deaerators suppresses the process of temperature increase. However, approximately five hours after the beginning of the accident the temperatures start to increase once again. This means that it is necessary to resume water supply to the GDH via non-regular means (*e. g.*, fire machine). The use of non-regular means with a low-pressure water source is possible, because the pressure in MCC is low. The acceptance criterion for the fuel cladding (700 °C) in the average loaded channels is reached within approximately 18600 s (more than five hours) after the beginning of the accident.

Because the de-pressurisation of MCC and the subsequent water supply from regular and non-regular means to the GDH in the case of loss of long-term cooling gives considerably better results compared with the other two measures (decay heat removal from the core by ventilation of DS compartments and by direct water supply into the reactor cavity), this way of accident management is recommended to be included in the RBMK-1500 accident management programme.

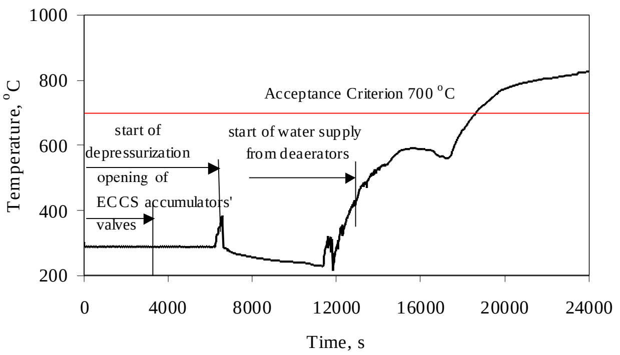

Fig. 8. Analysis of station blackout with de-pressurization of MCC and water supply using non-regular means. Peak cladding temperature in the average loaded channel

7. CONCLUSIONS

Results of Level 1 PSA of the Ignalina NPP have shown that in risk topography, the failure of the core long-term cooling is the main contributor to the frequency of the core damage. However, the transition to the condition of the reactor core due to a loss of long-term cooling specifies potential opportunities for the management of the accident consequences.

The most likely initiating event, which probably leads to the loss of long-term cooling, is station blackout. Analysis of the station blackout was performed using the RELAP5 model of Ignalina NPP reactor primary circuit and plant safety systems. Three ways of potential accident management for loss of the long-term core cooling were discussed:

- decay heat removal from the core by ventilation of DS compartments,
- decay heat removal from the core by direct water supply into the reactor cavity,
- de-pressurisation of the reactor coolant system and water supply from ECCS hydro-accumulators, deaerators or using non-regular means to the GDH.

The results showed that the first two ways are inexpedient. The ventilation of DS compartments and direct water supply into the RC are not sufficient to remove decay heat from the core. However, the de-pressurisation of MCC enables to mitigate the consequences of the loss of long-term core cooling. Therefore, this way of the mitigation of accident consequences is recommended to be included into the RBMK-1500 accident management programme.

Nomenclature

ALS	Accident Localisation System
CPS	Control and Protection System
DS	Drum Separator
ECCS	Emergency Core Cooling System
FC	Fuel Channel
GDH	Group Distribution Header
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
MFWP	Main Feed Water Pump
MSV	Main Safety Valve
NPP	Nuclear Power Plant
PSA	Probabilistic Safety Assessment
RBMK	Russian Acronym for “Channeled Large Power Reactor”
RC	Reactor Cavity
TCV	Turbine Control Valve

Received
23 January, 2003

References

1. Almenas K., Kaliatka A., Uspuras E. Ignalina RBMK-1500. A source book. Extended and updated version. Lithuanian Energy Institute, Kaunas, 1998. 198 p.
2. The RELAP5 Code Development Team. RELAP5/MOD3 Code Manual. Volume 1. Code Structure, System Models, and Solution Methods. NUREG/CR-5535, INEL-95/0174, 1995.
3. Kaliatka A., Uspuras E. Benchmark analysis of main circulation pump trip events at the Ignalina NPP using RELAP5 code, Nuclear Engineering and Design. Vol. 202. No. 1. 2000. Elsevier Science S. A. P. 109–118.
4. Uspuras E., Kaliatka A. Relap-based safety analysis of the Ignalina NPP. Proceedings of the Topical Meeting on Safety of Operating Reactors, 1995, Seattle. P. 307–313.
5. Vilemas J., Uspuras E., Kaliatka A. Benchmark analysis of inadvertent ECCS actuation using RELAP5 and ATHLET codes. Proceedings of the 6th International Conference on Nuclear Engineering. ICONE-6355. 1998.

Algirdas Kaliatka

AVARIJŲ VALDYMAS RBMK-1500 NETEKUS ILGALAIKIO AKTYVIOSIOS ZONOS AUŠINIMO

S a n t r a u k a

Ignalinos AE pirmojo lygio tikimybinis saugos įvertinimas parodė, kad rizikos topografijoje vyrauja pereinamieji procesai, o ne avarijos su šilumnešio praradimu, ir ilgalaikis aktyviosios zonos aušinimo netekimas yra pagrindinis veiksnys, turintis įtakos aktyviosios zonos pažeidimo dažniui. Tačiau pereinamieji procesai, vykstantys aktyviojoje zonoje ilgalaikio aušinimo netekimo atveju, įgalina numatyti priemones šių avarijų pasekmėms valdyti.

Šiame darbe pateikta išsami termohidraulinė ilgalaikio aktyviosios zonos aušinimo netekimo analizė, atlikta naudojantis RELAP5 programų paketo pagalba sukurtu Ignalinos AE reaktoriaus cirkuliacijos kontūro ir saugos sistemų modeliu. Šios analizės pagrindu sukurta tokių avarijų valdymo strategija.

Raktažodžiai: RBMK-1500, ilgalaikis aktyviosios zonos aušinimo praradimas, avarijų valdymo strategija

Альгирдас Калятка

УПРАВЛЕНИЕ АВАРИЯМИ В РБМК-1500 ПРИ ПОТЕРИ ДОЛГОВРЕМЕННОГО ОХЛАЖДЕНИЯ АКТИВНОЙ ЗОНЫ

Р е з ю м е

Результаты первого уровня вероятностного анализа безопасности (ВАБ) в Игналинской АЭС показали, что в топографии риска переходные процессы доминируют над авариями с потерей теплоносителя. ВАБ показал, что потеря долговременного охлаждения активной зоны является основным фактором, влияющим на частоту ее повреждения. Однако переходные процессы в активной зоне при потери долговременного охлаждения позволяют предусматривать возможности управлять последствиями этих аварий.

В настоящей работе представлен детальный термогидравлический анализ потери долговременного охлаждения активной зоны, выполненный с помощью модели контура охлаждения реактора и систем безопасности Игналинской АЭС, созданной на базе программного пакета RELAP5. На основе этого анализа развита стратегия управления такими авариями.

Ключевые слова: РБМК-1500, аварии с потерей долговременного теплоотвода, стратегия управления авариями