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IGNALINA NUCLEAR POWER PLANT SAFETY ISSUES

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ABSTRACT

The Ignalina Nuclear Power Plant (NPP) consists of two units, commissioned in 1983 and 1987. Unit 1 of Ignalina NPP was shutdown for decommissioning at the end of 2004 and Unit 2 is to be operated until the end of 2009. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500. The safety issues related with fulfilment of two fundamental safety functions – control of the reactivity and confinement of radioactive materials are addressed in this paper. The main systems, performing these safety functions at NPP-s, are Reactor Shutdown and Containment systems. The containment function at the Ignalina NPP is performed by the Accident localisation system. The performance and safety justification of Ignalina NPP additional and diverse shutdown systems, as well as of Accident localisation system, is discussed in this paper. The performed safety evaluations demonstrate that Ignalina NPP has adequate safety systems for reactor shutdown and confinement of radioactive materials.

KEYWORDS

Nuclear power plant, Safety analysis, Fuel channel, RBMK-1500.

INTRODUCTION

Preparatory works of construction of the Ignalina Nuclear Power Plant (NPP) have been started in 1974, and the first unit of Ignalina NPP was commissioned at the end of 1983, the second unit – in 1987. Nowadays because of political reasons the first unit of Ignalina NPP is shutdown, the second unit is planned to shutdown at the end of 2009.

Ignalina NPP with RBMK-1500 reactors belongs to the second generation of RBMK type reactors (it means, that this is most advanced version of RBMK reactor design series in comparison with others RBMK type NPP-s). In comparison with infamous Chernobyl NPP, Ignalina NPP reactors are by a third more powerfully and already from the beginning of operation had substantially advanced emergency protection systems (e.g., emergency core cooling and accident localization systems) [1].

The first safety justification (TOB) of Ignalina NPP has been prepared in 1989 by experts of NIKIET - designer and developer of RBMK reactors. The accident analysis for TOB was performed using at that time existing tool – quasistationary derivative approximation method, being based on conservative assumptions and existing experimental data. From the present-day viewpoint such safety justification [2] has lacks because it was limited only to the systems description and the analysis of design basis accidents. Computer codes, developed in Russia, have been used for simulations. These codes have not been verified as well the independent expertise of safety analysis has not been performed.

Therefore, at the beginning of the 90-s of the last century reasonably there were doubts how such safety justification of Ignalina NPP, presented in TOB, corresponds to the real situation. In 1992 at G7 Munich Summit the decision on closing of Soviet-design nuclear power plants, first of all the NPP-s with reactors of RBMK and VVER-440/230 types was accepted. In 1994 Lithuania has signed the agreement with the EBRD Account of Nuclear Safety on which has undertaken to perform in-depth safety analysis of the Ignalina NPP and to not change fuel channels in a reactor.

Right from the start, when Lithuania assumed control of the Ignalina NPP in 1991, the plant, its design and operational data has been completely open and accessible to Western experts. A large number of international and local studies have been conducted to verify the operational characteristics of the Ignalina NPP and analyse its level of risk. As a consequence, the information and conclusions regarding the safety criteria of the INPP stem not from a single group of analysts, but are based on a large number of very exhaustive studies by international experts having significant nuclear expertise.

In 1995 – 1996 very extensive Safety analysis report was developed employing US and Western Europe methodology and computer codes [3]. The purpose of this international study was to provide a comprehensive overview of plant status with special emphasis placed on its safety aspects. Specialists from the Ignalina NPP, Russia (the RDIPE institute, the main designer of channel type reactors), Canada and Sweden contributed.

One of the basic conclusions in this safety analysis report was such that in this case there was no problem, which would demand immediate shutdown of the Ignalina NPP.

In parallel with the Ignalina NPP Unit 1 safety analysis report in 1995–1997 it was performed its independent expertise and "Review of the Ignalina NPP Safety Anglysis Report" [4] was issued. This study was performed by experts from USA, Great Britain, France, Germany, Italy, Russia and Lithuania. Independent review has confirmed the main conclusions of Safety Analysis Report.

MODIFICATIONS OF REACTOR SHUTDOWN SYSTEM AND ITS SAFETY JUSTIFICATION

In the Ignalina NPP Unit 1 safety analysis report [3] have been investigated not only basic design accidents (discussed above), but also anticipated transients without scram (ATWS). Investigations of such accidents are carried out at the licensing process for USA and Western Europe nuclear power plants, however for the NPP-s with RBMK type reactors such analysis has been performed for the first time. The purpose of the analysis of ATWS was to define necessity of second, diverse reactor shutdown system, to estimate available time for mitigation of accident consequences and to take the first step for implementation of the concept on accident management. Consequences of accident for RBMK-1500 reactor, during which loss of preferred electrical power supply and failure of automatic reactor scram occurs [5], are presented in Figure 1. Due to loss of preferred electrical power supply all pumps are switched off therefore the coolant circulation through fuel channels is terminated. Because of the lost circulation fuel channels are not cooled sufficiently therefore temperature of the fuel channels walls starts to increase sharply. As it is seen from Figure 1 (a), already after 40 seconds from the beginning of the accident the peak FC wall temperature in the high power channels reaches acceptance criterion 650 ^{o}C. It means that because of the further increase of temperature in fuel channels plastic deformations begin – the channels because of influence of internal pressure can be ballooned and ruptured. On the 1-st second of accident the main electrical generators and turbines are switched off as well. Steam generated in the core is discharged through the steam discharge valves; however their capacity is not sufficient Therefore the pressure in reactor cooling circuit increases and approximately after 80 seconds from the beginning of accident reaches acceptance criterion 10.4 MPa (Figure 1, b). The further increase of pressure can lead to rupture of pipelines.

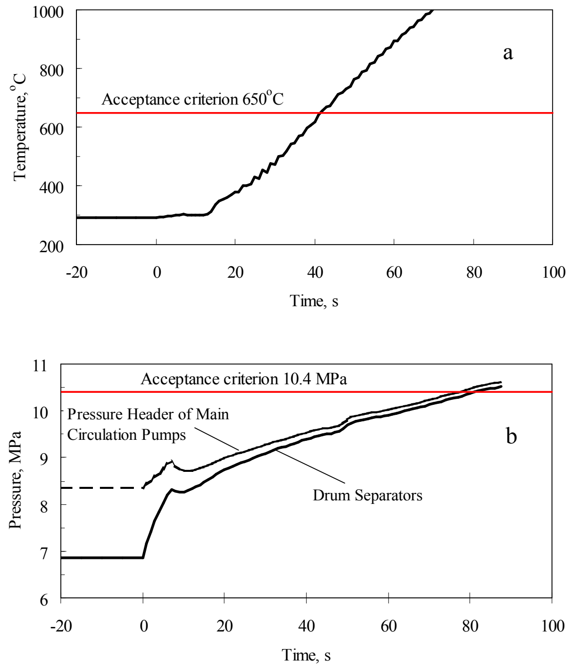

Figure 1. Analysis of loss of preferred electrical power supply and simultaneous failure of automatic reactor scram: a) the peak FC wall temperature in the high power channel, b) pressure behaviour in reactor cooling circuit

Thus the analysis of ATWS has shown that in some cases the consequences can be dramatic enough. Therefore the priority recommendation has been formulated: to implement the second diverse shutdown system, which shall use different (from existing system) principles for its operation. However development, designing and implementation of such system needed few years (in the Ignalina NPP unit 2 this system was installed in 2004), so the temporary system „**DAZ**" („Dopolnitelnaja avarijnaja zashchita" – „Additional emergency protection") has been developed and implemented. DAZ system used the same control rods as well as design reactor shutdown system, however signals for this system control were generated independently in respect of design reactor shutdown system. The scientists of the Lithuanian energy institute selected set points for DAZ activation and developed the safety justification for this system. Performed analysis has shown, that after implementation of DAZ system the reactor is shutdown in time, cooled reliably and acceptance criteria are not violated even in case of transients when design reactor shutdown system fails. In Figure 2 is shown the behaviour of the main parameters of reactor cooling circuit in case of loss of preferred electrical power supply and simultaneous failure of design reactor scram system. In this case two signals for operation of DAZ system (reactor shutdown) are generated: on increase of pressure in drum – separators and on decrease in the coolant flow rate through the main circulation pumps. In Unit 1 DAZ system was installed in 1999, in Unit 2 – 2000.

The Diverse Shutdown System (DSS) has been designed and installed in Ignalina NPP Unit 2 in 2004. In the first unit of Ignalina NPP this system has not been installed because reactor has been shutdown in 2004. Therefore, nowadays Ignalina NPP reactor emergency shutdown system consists of two independent shutdown systems: first – (BSM) controls manual control rods and shortened absorber rods, which are inserted into the core from bottom. This system performs the normal reactor shutdown function and can maintain a reactor in sub-critical state. Second system (AZ) controls 24 fast acting reactor scram rods as well additionally 49 rods, which belong both – BSM and AZ systems. AZ system performs emergency shutdown function. Also the Additional Hold-down System of the reactor is installed. This system allows to prepare and inject water and neutron absorber gadolinium mixture into control rods cooling circuit. Thus the reactor remains in subcritical state even in the case of failure of BSM system.

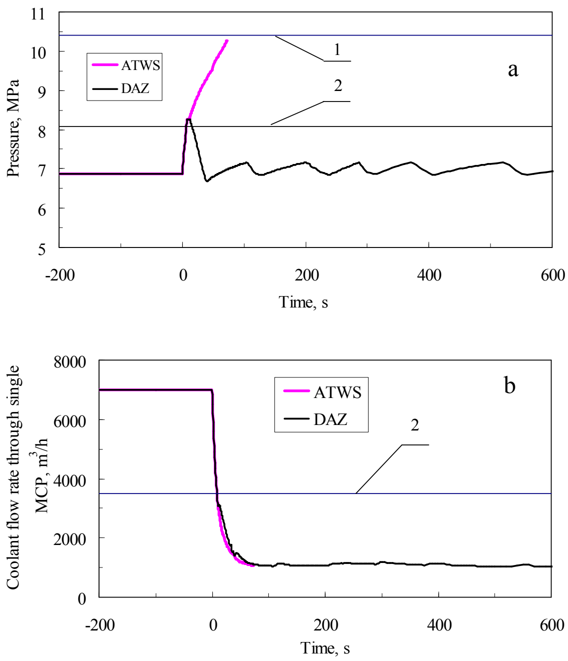

Figure 2. Analysis of loss of preferred electrical power supply and simultaneous failure of design reactor scram system, when DAZ system was installed: a) pressure behaviour in drum - separators, b) coolant flow rate through one main circulation pump, 1 – acceptance criterion, 2 – set points of DAZ system activation (reactor shutdown)

DSS justification was one of the main projects increasing a level of NPP safety. In this successfully finished PHARE financed project LEI employees together with experts from the countries of the Western Europe checked and have assessed the design documentation, carried out independent calculations thus helping regulatory body VATESI to make the decisions concerning implementation of mentioned system at Ignalina NPP [6]. During work the Ignalina NPP Unit 2 reactor models have been developed considering both shutdown systems. ATHLET and QUABOX/CUBBOX computer codes developed by Germany GRS mbH company were used for these models. Results of review calculations have confirmed reliability and validity of suggested reactor shutdown system. In conclusions of expertise it has been shown, that implementation of second, diverse reactor shutdown system protects a reactor in case of failure of design reactor scram system. Implementation of this system has ensured that any initiating event cannot cause accident with damage of the reactor core, as well as decreases reactor core damage frequency from $4 \cdot 10^{-4}$ up to $5 \cdot 10^{-6}$. According to the international requirements this parameter for the operating nuclear power plants should not exceed 10^{-4} per year, and for new NPPs, which are in process of construction – 10^{-5}. Therefore Ignalina NPP fulfils this criteria.

CONTAINMENT ISSUE

The function of a 'containment' is to ensure that in the unlikely event that radioactive materials are released from a fuel element, these materials do not escape to the environment. In many (though not all) Western reactors this is accomplished by a prominently visible hemispherical structure. In the Ignalina NPP the function of containing accidentally released radioactive material is accomplished by an extensive system of interconnected steel lined, reenforced concrete compartments called the ALS (Accident Localization System). The functional principles of the ALS are shown schematically in Figure 3.

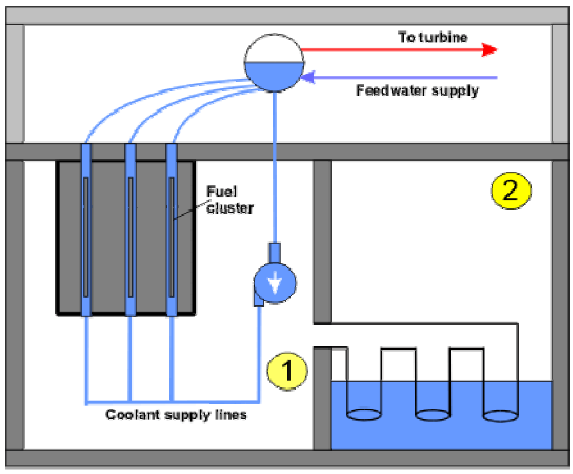

Figure 3. Functional Schematic of the Ignalina NPP Accident Localization System

The ALS uses the 'pressure suppression' principle employed by G.E. designed boiling water reactors. This approach divides the space around the reactor core and its piping into two volumes which are connected by submerged path-ways underneath large internal water pools In the unlikely event that the primary system is breached, the emitted, potentially radioactive steam raises the pressure within the internal compartment (Figure 3, Vol 1), this compresses the water level downward in the submerged vents and the radioactive steam is forced to bubble through the internal water pools. This condenses the vapor, lowering the pressure and also dissolves most of the radioactive materials. The outer compartments (Figure 3, Vol 2) then contain the residual uncondensed materials.

The Ignalina NPP ALS encloses the large reactor core, the coolant pumps and all of the piping providing coolant to the core. It in not necessary to enclose the pipes above the reactor core, which carry the exiting two-phase (steam-water) mixture to the drum separators, because if one of them is breached, coolant flow to the fuel channels (which is provided by pipes entering the core from bellow) will not be interrupted. Significant amounts of radioactive material can escape only if fuel clusters are over-heated. Breaches in the exiting pipes will not reduce coolant flow, therefore the fuel elements will not overheat.

The effectiveness of the ALS has been verified by extensive international analysis and experimental programs [3], [4], [7]. They all show that even if events leading to release of radioactive materials are postulated, these materials will be contained by the ALS. The minute amounts (due primarily to non-condensable noble gases) which would eventually reach the environment, would not exceed the amounts that would be released by Western Europe and US built reactors provided with the more familiar, prominently visible 'dome containments'.

One of possible accidents, when ALS activation is required, is single or multiple fuel channel rupture within reactor cavity. Below is presented the assessment of Ignalina NPP RBMK -1500 capability to withstand multiple fuel channel tube ruptures.

MULTIPLE FUEL CHANNEL RUPTURES

In case of fuel channel rupture a two-phase flow is discharged to the reactor cavity (RC). The additional pressure increase in RC appears due to vaporisation of liquid on the hot surfaces of cavity structures i.e. graphite and FC. The RC is protected against overpressure by the reactor cavity venting system (RCVS), which directs the released coolant from RC to the condensing pools of ALS.

In the case of multiple fuel channel tube ruptures, if the RCVS does not assure relief of steam-water-gas mixture from RC, the pressure increase in the RC will lift top plate of the RC. Those, structural integrity of the RC and remaining FC will be lost. Therefore it is important to maintain RC integrity, which is assured if pressure in the RC is below permissible pressure (314 kPa, abs).

The initial design assured the integrity of RC in the case of up to 3 FC ruptures. In 1996 the modernization of the RCVS was implemented, which in accordance with design calculations of modernization assured the integrity of RC in the case of up to 9 FC ruptures. However, these design calculations are conservative and involve a lot of uncertainties.

The best estimate analysis of Ignalina NPP response to Multiple pressure tubes (i.e. FC) rupture was performed at the Lithuanian Energy Institute [8]. Calculation of coolant mass and energy release to the RC in case of FC rupture was performed using the main circulation circuit model of Ignalina NPP, which was developed by employing state-of-the-art code RELAP5/Mod3.2 [9]. These results were applied further for the analysis of the thermal hydraulic parameters behaviour in the RCVS and ALS employing CONTAIN code [10]. The uncertainty and sensitivity analysis was performed by employing the code SUSA 3.2 [11]. The uncertainty and sensitivity analysis showed that in case of 11 FC rupture the limiting pressure in the RC (314 kPa) was not exceeded even in most unfavorable combinations of the initial conditions and modeling parameters (Figure 4).

Summarizing the results of the uncertainty and sensitivity analysis, it was concluded, that the capacity of RCVS comprises from 11 up to 19 FC, i.e. 15±4 FC.

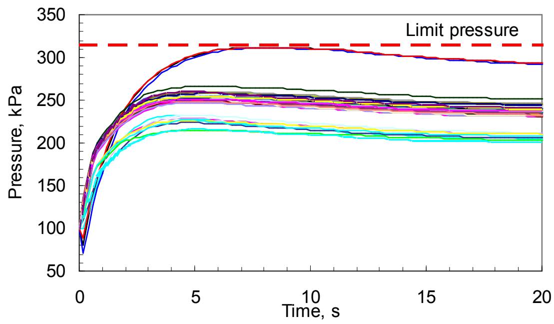

Figure 4. Pressure in the reactor cavity in the case of 11 FC rupture

CONCLUSIONS

Many international studies have been performed to investigate and assess safety level of the Ignalina NPP. Design parameters of the Ignalina NPP as well level of its risk have been investigated during these studies. Design and operational data has been completely open and accessible to Western experts.

On the basis of the performed investigations, efforts of local and international experts had been developed recommendations on improvement of the plant safety. These measures have allowed to improve constantly safety level of Ignalina NPP.

The performed safety evaluations demonstrate that Ignalina NPP has adequate safety systems for reactor shutdown and confinement of radioactive materials and that these systems fulfill the safety functions in accordance with Lithuanian and International requirements for safety of nuclear power plants. Now there are no significant differences in safety terms, comparing the RBMK-1500 of the Ignalina NPP and other NPP-s of the same generation, constructed and operating in Western Europe or US.

REFERENCES

[1] Almenas K., Kaliatka A., Uspuras E. (1998). Ignalina RBMK-1500. A Source Book. Extended and Updated Version. Lithuanian Energy Institute, Kaunas, Lithuania.

[2] Отчет НИКЕЭТ № Д040-1110 (1989) Техническое обоснование безопасности реакторной установки первой очереди ИАЭС. Москва, 1989.

[3] Ignalina NPP, 1996. In-depth safety assessment of Ignalina Nuclear Power Plant. Ignalina NPP report, Visaginas, Lithuania.

[4] RISKAUDIT Report No 55 (1997) Review of the Ignalina Nuclear Power Plant Safety Analysis Report, Summary. RISKAUDIT. Paris, France, 1997.

[5] Kaliatka A., Ušpuras E. (2002) Development and evaluation of additional shutdown system at the Ignalina NPP by employing RELAP5 code // *Nuclear Engineering and Design*. ISSN 0029-5493. Vol. 217. P. 129–139.

[6] LEI Report (2004) Independent review calculations of the accidents for Ignalina NPP Unit 2 DSS safety substantiation. LEI report according to project „TSO Support to VATESI during Review and Licensing of the Diverse Shutdown System at Ignalina NPP Unit 2“. LEI, Kaunas, 2004.

[7] Almenas K., Cesna B., Kaliatka A. and other, (1999). Thermal-hydraulic Evaluation of the RBMK-1500 Accident Confinement System using CONTAIN 11AF. Nucl. Eng. Des., 191/1, 83-99.

[8] Čėsna B., Rimkevičius S., Urbonavičius E., Babilas E. (2004) Reactor cavity and ALS thermal-hydraulic evaluation in case of fuel channels ruptures at Ignalina NPP // *Nuclear Engineering and Design*. ISSN 0029-5493. 2004. Vol. 232. P. 57–73.

[9] Fletcher C. D., et. al. (1992). RELAP5/MOD3 code manual user's guidelines. Idaho National Engineering Lab., NUREG/CR-5535.

[10] Murata K. K. (1995). Change document for update C11AF: CONTAIN 1.2 pre-release bugfixes.

[11] Kloos M., Hofer E. (1999). SUSA Version 3.2. User's Guide and Tutorial.