Modelling of Thermal HydraulicTransient Processes in Ignalina NPP Compartments

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Modelling of Thermal Hydraulic Transient Processes in Nuclear Power Plants: Ignalina Compartments

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PREFACE

The Ignalina Nuclear Power Plant is Lithuania's only nuclear power plant producing ~80 % of electricity in the country. The plant consists of two units with RBMK 1500 reactors commissioned in December 1983 and August 1987. The reactor of Unit 1 is shutdown in December of 2004.

The RBMK-1500 is a graphite moderated, boiling water, channel type reactor with a total of 1661 vertical parallel fuel channels and numerous components such as headers, pumps, valves, etc. The RBMK-1500 reactors of the Ignalina NPP are protected by a pressure suppression type containment, which, because of its specialized nature, is referred to as the Accident Localisation System (ALS). The ALS forms the last barrier preventing radioactive material release to the environment, i.e. it is designed to prevent the release of contaminated steam-water mixture to the environment in case of Loss-of-Coolant Accidents (LOCA). The ALS response to the range of LOCA is described in this monograph. One of the peculiarities of Ignalina NPP with RBMK-1500 reactors is that not all of the reactor coolant circuit is enclosed within ALS. In this case the function of last barrier perform other building structures of Ignalina NPP. Nevertheless, the requirements of safety standards regarding limits of radiological doses shall be met in the case of any LOCA. Therefore, besides the ALS response analysis, other possible places of LOCA are considered in this monograph, and the analysis of thermal hydraulic parameters in the affected by LOCA Ignalina NPP compartments is performed.

Many individual and several cooperative projects have greatly expanded the design data base and the general understanding of RBMK plants in the last few years. Notable in this respect are Safety Analysis Reports, Barselina project, Probabilistic Safety Analysis of Level 2, and number of the reports issued in the frames of PHARE and European Union Framework Programme as well as in the frames of bilateral co-operation between LEI and foreign organisations (for example, GRS mbH (Germany), Jacobsen Engineering Ltd (UK) and Scientech (USA). The study presented in this monograph fits into the above context. It serves both to summarise the information available regarding the unique system of Ignalina NPP compartments and it employs state-of the-art analysis techniques to verify the response of the ALS and other compartments to a broad range of LOCA events.

The authors of this monograph would like to acknowledge the prof. K. Almenas, who is one of initiators of safety analysis activities in Lithuanian Energy Institute and who provided the first lessons to LEI scientists applying the computer codes (incl. CONTAIN) for the NPP safety analysis and especially for Ignalina NPP.

The authors of this monograph would like to acknowledge the access to the RALOC4 and COCOSYS code versions provided by the GRS. We are thankful to GRS experts Mr. H. Wolff and Mr. S. Arndt for collaboration and support performing analytical investigations by employing the RALOC4 code. We also want to extend our thanks to the administration and technical staff of Ignalina NPP, for providing information regarding the operational procedures and the operational data. The authors acknowledge the scientists of Lithuanian Energy Institute Dr. E. Urbonavičius and Dr. Habil. B. Čėsna for contribution performing a plenty of various calculations of thermal hydraulic parameters behaviour in Ignalina NPP compartments.

Authors

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NOMENCLATURE AND ABBREVIATIONS

A	–	flow path area, m^2;
d	–	diameter of pipeline, m;
F_P	–	pool area, m^2;
h	–	zone elevation, m;
I	–	enthalpy, J;
i	–	specific enthalpy, J/kg;
l	–	length of pipe Section, m;
L	–	length of the flow path, m;
m	–	mass, kg;
p	–	gas pressure, Pa;
Q	–	energy, J;
r_{ex}	–	bubble expansion radius, m;
T	–	temperature, K;
t	–	time, s;
U	–	internal energy, J;
V	–	volume, m^3;
W	–	mass flow rate, kg/s;
Δp_{gr}	–	pressure head due to gravity, Pa;
λ	–	friction coefficient;
ξ	–	flow resistance coefficient for the flow path;
ξ_d	–	hydraulic resistance coefficient for pipeline;
ξ_{loc}	–	local hydraulic resistance coefficient;
ρ	–	density, kg/m^3.

Subscripts

i	–	component of non-condensable gases;
j	–	junction;
p	–	pool;
s	–	source zone;
t	–	target zone;
v	–	vapour;
w	–	water.

Abbreviations

ALS	–	Accident Localisation System;
ALT	–	Accident Localisation system Tower;
AVC	–	Air Venting Channel;
AZ	–	Reactor Emergency Protection System;
BSRC	–	Bottom Steam Reception Chamber;
BWR	–	Boiling Water Reactor;
CON	–	Condensation on structures;
CP	–	Condensing Pool;
CPS	–	Control and Protection System;
CTCS	–	Condenser Tray Cooling System;
CWL	–	Control of Water Level;
DS	–	Drum Separator;
ECCS	–	Emergency Core Cooling System;
EFWP	–	Emergency Feed Water Pump;
FASS	–	Fast Acting Scram System;
FC	–	Fuel Channel;
FOC	–	Forced convection model;
FP	–	Fission Products;
FRC	–	Free convection model;
GDC	–	Gas Delay Chamber;
GDH	–	Group Distribution Header;
GRC	–	Gas Delay Chamber;
GRS	–	Gesellschaft für Reaktorsicherheit (Germany society on safety of installations and reactors);
HCC	–	Hot Condensate Chamber;
HEPA	–	High Efficiency Particulate Air;
IAEA	–	International Atomic Energy Agency;
IRV	–	Isolating and Regulating Valve;
LBB	–	Leak Before Break;
LDS	–	Leak Detection System;
LOCA	–	Loss of Coolant Accident;
LWP	–	Lower Water Piping;
MCC	–	Main Circulation Circuit;
MCP	–	Main Circulation Pump;
MDBA	–	Maximum Design Basis Accident;
MER	–	Mass/energy release;
MFWP	–	Main Feed Water Pump;
MSD	–	Membrane-Safety Device;

MSIV	–	Main Steam Isolation Valve;
MSRV	–	Main Steam Relief Valve (sometimes called also as MSV);
MSV	–	Main Safety Valve;
NPP	–	Nuclear Power Plant;
PBB	–	Reinforced Leaktight Compartment;
PH	–	Pressure Header;
PSA	–	Probabilistic Safety Analysis;
PSS	–	Pressure Suppression System;
PWR	–	Pressure Water Reactor;
RBMK	–	Russian acronym for ‘Large Channel Type Water Cooled Graphite Moderated Reactor’;
RC	–	Reactor Cavity;
RCVS	–	Reactor Cavity Venting System;
RDIPE	–	Research and Development Institute of Power Engineering;
RLC	–	Reinforced Leaktight Compartment;
RSR	–	Review of Safety Analysis Report;
SAR	–	Safety Analysis Report;
SDD	–	Steam Distribution Device;
SDV–A	–	Steam Discharge Valve to ALS tower;
SDV–C	–	Steam Discharge Valves to Condenser;
SDV–D	–	Steam Discharge Valve to the Deaerator;
SH	–	Suction Header;
TCV	–	Turbine Control Valve;
TOB	–	Russian acronym of ‘Safety Justification Report’;
TSRC	–	Top Steam Reception Chamber;
URC	–	Under Reactor Compartment;
USA	–	United States of America;
VNIPIET	–	Russian acronym for ‘All-Union Research and Development Institute for Energy Technology’.

1
Introduction. Main features of NPP containments

The Ignalina nuclear power plant (INPP) reactor RBMK-1500 is a graphite moderated, boiling water, channel type reactor. The reactor fuel elements are enclosed in individual, pressurized flow channels. INPP reactor core has 1661 such channels. Both the reactor and a large portion of the cooling circuit are enclosed in reinforced protective shells of Accident localisation system. The function of ALS is to ensure that in the unlikely event when radioactive materials are released from a fuel element, these materials do not escape to the environment.

A characteristic feature of nuclear power plants is the containment, or protective shell. This is a large, especially strong, steel and reinforced concrete building, usually semi cylindrical in shape, which encloses the reactor and its cooling circuits. The Ignalina NPP does not have such a containment structure. Almost all components of the power plant are located in large, interconnected, traditional buildings. This external image of the power plant has contributed to the widespread opinion that the Ignalina NPP has no containment system. That is not true. In fact, both the reactor and a large portion of the cooling circuit are enclosed in reinforced protective shells. In addition, a complicated large scale system is provided, purpose of which is to condense the steam erupting from the cooling circuit during a possible accident. This equipment takes up a significant fraction of the volume of the central power plant building. The INPP is cooled by high pressure water. While the reactor is on line, most of thermal energy is found, not in the core, but in the cooling circuit. Therefore, if an accident occurs which damages the cooling circuit (pipe break, valve stuck open, etc.), the main function of the containment system is to prevent the released high pressure steam-water mixture from reaching the atmosphere. Generally, there are two containment types, which achieve this result: 1) dry containment; 2) pressure suppression containment. The system used at the Ignalina NPP belongs to the second category of containment.

1.1 Containment design types

The most common type is called 'dry containment' (Figure 1.1). In the case of this type containment, the primary containment envelope is a steel shell or a concrete building (cylindrical or spherical) with a steel liner that surrounds the nuclear steam supply system. The containment encompasses all components of the reactor coolant system under primary pressure. It is designed as a full pressure containment; i.e. it is able to withstand the increases in pressure and temperature that occur in the event of any design basis accident, especially a LOCA. This type of containment can withstand the pressure reached when the entire contents of the cooling circuit is suddenly lost from the circuit (within half a minute). Containment buildings are correspondingly large and expensive.

Energy management in the building of dry containment can be accomplished by an air cooler system or by a water spray system. In addition, the free volume of the containment and the structural heat sinks (the containment envelope and the structures within it) are used to limit peak pressures and temperatures in postulated conditions for pipe rupture accidents. The initial supply of water for the spray system and for the emergency core cooling system (ECCS) is held in a large tank. When this water has been used, suction for both the spray system and the emergency core cooling system is switched to the containment building sump. Water, which is recirculated to the reactor vessel is sometimes cooled by means of heat exchangers. In most designs the recirculation water for the spray headers – which is also used to limit contamination of the containment atmosphere – is cooled by means of heat exchangers. When pipes rupture in systems other than the reactor coolant system, only the spray system is operated in the recirculation mode. There exist several modifications of dry containments (full pressure double wall containment, negative pressure containment system for a pressurized heavy water reactor) [1].

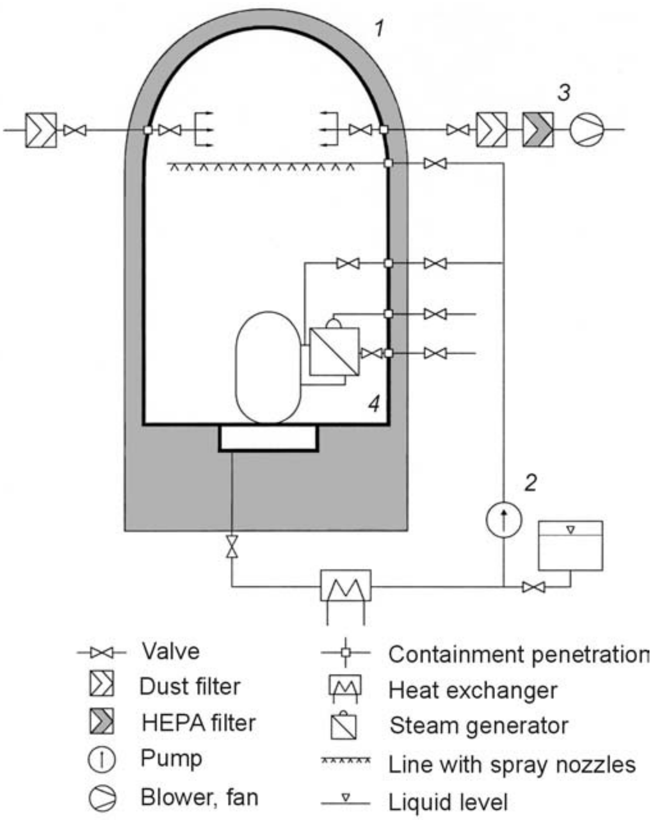

FIGURE 1.1 Schematic diagram of a full pressure dry containment system for a pressurized water reactor [1]: *1*, containment; *2*, containment spray system; *3*, filtered air discharge system; *4*, liner

The ‘Pressure suppression’ containment type uses additional equipment, which condenses a part of the released steam to reduce the peak pressures, which can be reached during an accident. The ‘Pressure suppression’ type containment could be split into three parts (Figure 1.2):

- dry well – compartments, where the reactor cooling system pipelines, devices and the reactor are installed. The radioactive coolant will be released in these compartments during the accident;

- condensing pool – water pool or ice condenser, which condense steam;
- wet well – compartments behind condensing pool. In this compartments the air and non-condensable gases are released after bubbling through pools water.

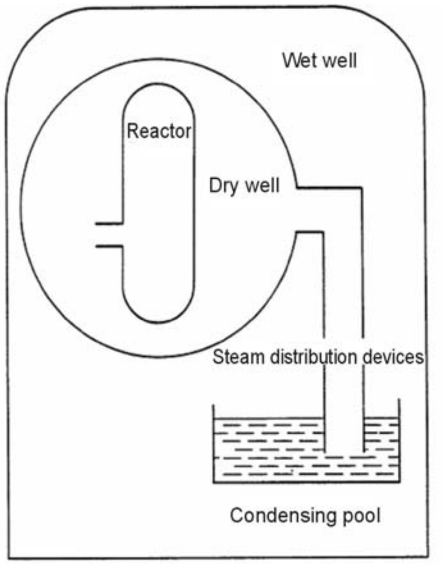

FIGURE 1.2 Schematic diagram of a 'Pressure suppression' containment [2]

During the accident in the NPP with 'Pressure suppression' type containment the steam released in the dry well is directed to the condensing pool. There the steam is condensed and non-condensable gases, such as air, go to the wet well and are delayed there.

At the moment in the World the three types of 'pressure suppression' containments and theirs modification are used (Figure 1.3). The first was developed Mark I type containment whit 'bulb' form of dry well, torus form of wet well and complex system of steam distribution devices. In the second and third – Mark II and Mark III – type containments the dry well was expanded. The Mark II type containment has not torus form compartments and the condensing pools are located below compartments of reactor cooling system pipelines. The long (~7 m), vertical inserted under the water pipes directs released steam to the condensing pool. The Mark III type containment has significant differences from Mark I and Mark II types of containments:

- the bound of dry well is not the part of the whole containment bound;
- openings, which connect dry and wet wells are horizontal and inserted under the water layer in condensing pool;
- volume of wet well is significantly larger than dry well volume;
- wet well is serviceable during reactor operation.

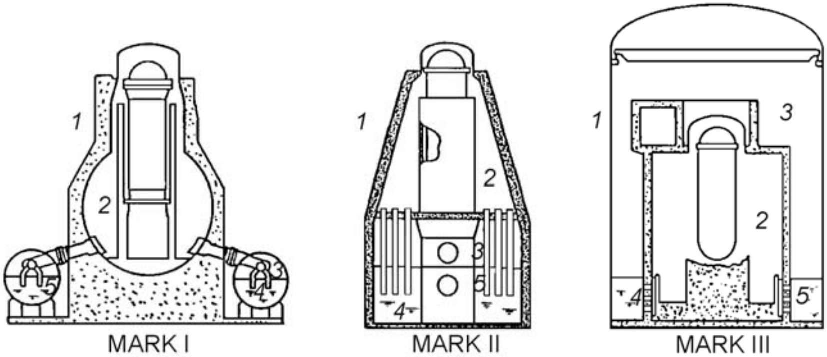

FIGURE 1.3 ‘Pressure suppression’ type containments evolution [3]: *1*, containment; *2*, dry well; *3*, wet well; *4*, condensing pools; *5*, steam distribution devices

To increase the steam condensation rate and to scrub radioactive materials in the dry and wet wells the sprinklers, which spray the water droplets, are installed. To avoid the hydrogen detonation, during the accident, some of BWR containments (usually Mark I and Mark II types as well as Swedish BWR) are filled with nitrogen. Comparison of Ignalina NPP ALS with Western NPP containments is given in Section 1.3.

The Russian designed VVER 440/213 reactors, differently from VVER 440/230 and VVER 1000, have ‘Pressure suppression’ type containments. Today, more than 20 reactors of such type are in operation or under commissioning in Czech Republic, Finland, Hungary, Russia, Slovakia, and Ukraine. VVER 440/213 containment description and its comparison with Ignalina NPP ALS are given in Section 1.4.

The ice condenser containment in PWR uses a concept for the pressure suppression system in which the high pressure steam-air mixture, resulting from an accident conditions pipe rupture, is directed through vent doors into chambers, containing baskets filled with ice. The steam condenses onto the surface of the ice in the baskets.

1.2 Brief characterisation of Ignalina NPP ALS

Accident localisation system (ALS) at Ignalina NPP forms the last barrier against the release of radioactive materials to the environment. According to functional principle, the ALS could be attributed to 'pressure suppression' type containment. It means that ALS uses condensing pools, which condense the released steam in order to reduce the peak pressures that can be reached during any loss of coolant accident. The main safety function for ALS is to contain the radioactive materials, that are released from a fuel element in the course of accident, and to limit the releases within specified limits.

ALS is an extensive system of interconnected steel lined, re-enforced concrete compartments. The functional principles of the ALS are shown schematically in Figure 1.4. The 'pressure suppression' 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 (Volume 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 vapour, lowering the pressure and also dissolves most of the radioactive materials. The outer compartments (Volume 2) then contain the residual un-condensed materials.

The ALS encloses the large Ignalina NPP reactor core, the coolant pumps and all of the piping providing coolant to the core. It is 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 in this case.

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