LAPPEENRANTA UNIVERSITY OF TECHNOLOGY

LUT School of Energy Systems

Degree Program in Nuclear Energy Engineering

Dmitrii Chalyi

**Failure modes of passive decay heat removing safety systems of modern nuclear power plants**

Examiners: Professor D.Sc. (Tech.) Juhani Hyvärinen,

M. Sc. (Tech.) Otso-Pekka Kauppinen.

# ABSTRACT

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**Failure modes of passive decay heat removing safety systems of modern nuclear power plants**

Master's Thesis

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70 pages, 29 figures and 8 tables.

Examiners: Professor D.Sc. (Tech.) Juhani Hyvärinen, M.Sc. (Tech.) Otso-Pekka Kauppinen

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Keywords: passive safety systems, nuclear power plant, SPOT PG, SPOT ZO, AES-2006, failure modes, decay heat removal, natural circulation, TRACE.

The purpose of this master's thesis is to gain an understanding of passive safety systems' role in modern nuclear reactors projects and to research the failure modes of passive decay heat removal safety systems which use phenomenon of natural circulation. Another purpose is to identify the main physical principles and phenomena which are used to establish passive safety tools in nuclear power plants.

The work describes passive decay heat removal systems used in AES-2006 project and focuses on the behavior of SPOT PG system. The descriptions of the main large-scale research facilities of the passive safety systems of the AES-2006 power plant are also included.

The work contains the calculations of the SPOT PG system, which was modeled with thermal-hydraulic system code TRACE. The dimensions of the calculation model are set according to the dimensions of the real SPOT PG system. In these calculations three parameters are investigated as a function of decay heat power: the pressure of the system, the natural circulation mass flow rate around the closed loop, and the level of liquid in the downcomer. The purpose of the calculations is to test the ability of the SPOT PG system to remove the decay heat from the primary side of the nuclear reactor in case of failure of one, two, or three loops out of four.

The calculations show that three loops of the SPOT PG system have adequate capacity to provide the necessary level of safety.

In conclusion, the work supports the view that passive systems could be widely spread in modern nuclear projects.

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May, 2016.

## Table of Contents

## List of Figures

## List of Tables

## List of Symbols and Abbreviations

| | |
|---|---|
| 3D | Three-Dimensional |
| BDBA | Beyond Design Basis Accidents |
| BWR | Boiling Water Reactor |
| CCC | Containment Cooling Condenser |
| CMT | Core Make-up Tanks |
| CSNI | Committee on the Safety of Nuclear Installations |
| DBA | Design Basis Accidents |
| EC | Emergency Condensers |
| GCDS | Gravity Driven Cooling System |
| I&C | Instrumentation and Control |
| IAEA | International Atomic Energy Agency |
| ICS | Isolation Condenser System |
| KMS | Containment Experimental Installation |
| LCA | Life Cycle Assessment |
| LOCA | Loss of Coolant Accident |
| NPP | Nuclear Power Plant |
| NRC | Nuclear Regulatory Commission |
| PPPT | Passive Pressure Pulse Transmitters |
| PRHR | Passive Residual Heat Removal |
| PWR | Pressurized Water Reactor |
| RPV | Reactor Pressure Vessel |
| SG | Steam Generator |
| SNAP | Symbolic Nuclear Analysis Package |
| TRACE | TRAC/RELAP Advanced Computational Engine |
| WWER | Water-Water Energetic Reactor |

# 1. INTRODUCTION

Nowadays nuclear power produces 11% of the electricity on the planet. Generating capacities are expanding so as the number of countries who want to establish nuclear power stations [1]. In our economically unstable world today nuclear power companies are using new approaches in design of nuclear facilities and trying to modernize well-performed proven means in order to diminish the capital costs of the whole nuclear power plant (NPP).

In order to reach new economic trends designers turned their attention to the passive systems of safety. The reason lies in the fact that passive safety systems do not require energy sources and do not contain moving parts, except of valves, that need to be open to initiate the function of the system. Furthermore, they do not require control signals, nor actions of operating personnel. These safety systems are called natural circulation systems and use mainly gravity and convection to perform their functions. [2]

### Natural circulation phenomenon

The natural circulation in the circuit of a reactor or other devices is achieved without pumps or other active elements. Generally speaking, a natural circulation system includes a heat source and heat sink connected to each other with pipes and situated on a different height (the heat sink at the upper level). [2] Figure 1 presents a simple configuration of the open natural circulation system.

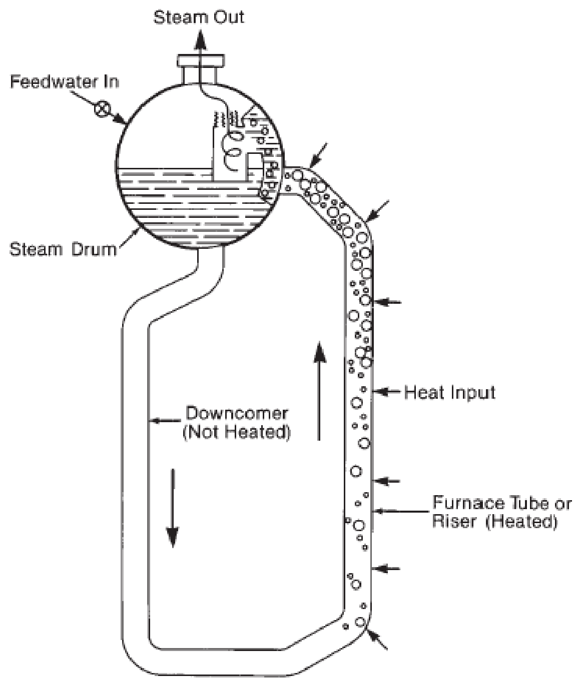


**Figure 1.** Natural circulation system. [3]

It can be seen from the figure that cold water in the downcomer is flowing downwards. The downcomer is not heated from external source. Therefore, there is no steam in this section. In the riser section water is heated with external heat source. Depending on the heating power and the system conditions, the heat input generates a steam-water mixture or increases the temperature of the water in riser section. Due to the fact that the mix of water and steam in the riser section is hotter and less dense than the water in the downcomer section, gravity will drive water to flow upwards in the riser section towards the steam drum. Generally speaking, the effect of natural circulation is achieved by the distinction in densities of the working fluid in the bottom of the system in question and in the top of it. [4]

Natural circulation phenomenon can exist in the following configurations [2]:

1. The source of heat and the heat sink of the primary loop are composed via lower elevated reactor and higher elevated steam generator (SG);
2. In the reactor pressure vessel (RPV) the natural circulation is formed between the reactor core and the downcomer. The steady state natural circulation between them arises because of a density difference of water between them. The temperature in the downcomer is lower than in the reactor core;
3. Closed loop cooling of the volume inside the facility's containment.

The natural circulation phenomenon is also used to take heat away from the reactor in normal operation conditions but this is not so common. One example of this kind of reactor is small reactor VK-50 located in Dimitrovgrad, Russia [5].

**Advantages and disadvantages of passive safety systems**

The natural circulation phenomenon has been researched extensively through the years and it was proved that the application of passive systems is desirable for NPPs [6],[7]. In accordance with the international atomic energy agency (IAEA) the advantages of passive systems/components outweigh the disadvantages, since the list of benefits is quite wide [8].

Using of the passive safety systems can improve the economics of the system design due to the simplification of the system [9]. The most important advantage of the passive systems is the lower construction, operation and maintenance costs. Application of passive systems reduces the number of components, and yield design simplifications, so that the number and complexities of safety actions can be reduced. Also the passive systems eliminate the need for instant actions of the operation personnel in case of an accident. The passive systems do not have moving parts and do not require actions of control system during the normal working regime and in case of the beyond design basis accidents (BDBA) and design basis accidents (DBA). The appliance of passive systems simplifies the whole structure of NPP, reduce the possibility of human errors, and provide increased time to avoid severe troubles in cases of an accidents. The passive systems based on the natural circulation do not require repair or maintenance work during their operation. It should not be forgotten that the actuation of the passive systems needs to have better reliability compared to traditional active systems providing the same function; otherwise the increase of the system reliability projected by implementation of the passive system may be lost. [10]

On the other hand, there are some serious disadvantages of the natural circulation systems. The most important disadvantages are the lower driving forces. In particular, in certain conditions where rapid actions are required, active systems may be more suitable to carry out certain safety functions. Also, a load follow operation may be limited in the reactors based on natural circulation moving of the primary side coolant. Therefore, in some new reactor designs originally designed for the natural circulation, the forced circulation flow (by water pumps) has been introduced to allow for the better load follow capability and to increase the reactor rated power. The scaling for the passive safety systems is more problematic in comparison with the scaling for the active ones. Therefore the application of an experimental or operational data acquired from a system with a size that differs from the system being designed may not be appropriate. The lower driving force might also require the use of larger equipment, which will reduce the cost savings obtained from the active systems' exclusion. [11] Moreover, larger parts might cause extra complications in a seismic characteristics of some units [12].

Thus, in general we can say that it is complicated to use only passive safety systems in nuclear power plant (NPP) projects, but their benefits should be used actively where it is possible. Passive features require computations with sophisticated analysis methods to assure that the systems will be able to perform their functions.

**Reliability of passive safety systems**

The reliability assessment of systems, which use natural forces in operation, depends on the environmental, physical, nuclear, or chemical phenomena, to a greater extent than active systems [6].

In IAEA general conference in 1991, the discussions about the safety operations of the future nuclear power plant designs were held. This meeting is the highest policy-making body of the organization [13]. In this session it was decided that wide usage of passive systems is appropriate because of their perspectiveness for future reactor designs [14].

The improved reliability of the passive systems compared to the active systems can be achieved not only due to the fact that the passive systems are generally simpler in design and therefore more reliable than the active ones, but also because the passivity of the system eliminates the need for the complex managing and supply systems (e.g. the power supply, the ventilation and air conditioning system, etc.), i.e. auxiliary systems that are needed for the active systems. In addition, these auxiliary systems are subjected to the various types of disturbances; the most harmful of which are fire, flooding, and erroneous actions of the personnel during the inspections and repairs of the system, and control process. [14]

**Future of passive safety systems**

It seems that new trends will result in new designs, which will promote a new stage of development of a nuclear power [7]. In the next generation of reactors the passive systems will be applied for stabilizing the operation of the reactor in the normal regimes of and for providing the cooling of the reactor following wider range of accidents than in the current designs [15].

Thus, the implementation of the natural circulation systems into the new nuclear power plant designs has been suggested to be the one of the main directions of the nuclear industry development.

**Thesis Purposes**

The main purpose of the thesis is to introduce a detailed review of the passive safety systems used in modern nuclear reactors projects and to carry out an analysis of passive heat decay removal aggregates used in AES-2006 project. The next step is to carry out a classification of failure modes of passive heat decay removal systems which performs with the use of natural circulation phenomenon. Third goal is to model one of the passive heat decay removal systems of AES-2006 power plant named SPOT PG with TRACE code and test it for effectiveness.

## 2. PASSIVE SAFETY SYSTEMS FOR DECAY HEAT REMOVAL ON ADVANCED NUCLEAR POWER PLANTS

The passive safety systems are being considered to provide an effective decay heat removal from the core and to deliver an additional stability for the nuclear reactor [15]. The important function of the passive safety systems is the cooling of the core during the DBA and BDBA. It is performed by the number of implements [14]:

- Adequate circulated flow of the coolant inside the system,
- Adequate coolant insertion in the system,
- Adequate heat transfer from the core,
- Ultimate heat sink provision.

The function named "ultimate heat sink" is a complex cooling water system which serves the plant during a variety of normal and emergency operating scenarios [16]. In current and advanced reactor concepts this feature is mainly performed using the water tanks which are located inside or outside the containment shell or using the surrounding air via heat exchangers. For example, in the WWER-1000/V-392 reactor concept the air heat exchangers located outside the containment act as the ultimate heat sink. [17]

During the various DBAs and BDBAs, a set of those implements listed above or even all of them may be required. To perform such a variety of functions plenty of passive systems are proposed for future reactor concepts. Nowadays the idea of entire water storage for replenishment of primary coolant inventory is common for many new concepts of NPPs. Such approach can improve protection against external events and reduce the risk of loss of coolant accidents with containment bypass. [17]

In addition, there are another ways developed to perform the function of replenishment of primary coolant inventory [10]:

1. Pressure relief via the relief tank to the water storage tank;
2. Taking away of heat from the primary circuit to the water tank using heat exchangers located inside the tank;
3. The combination of containment sump with water tank;
4. Water tank placed at higher elevation than the reactor core for gravity-driven injection;

5. Storage of a portion of water at high elevation under the full primary pressure for coolant injection at high pressure.

Coolant injection function is carefully developed in new concepts and almost all of them include a combination of different passive and active systems to provide it.

Broadly speaking, NPPs should maintain fulfillment of safety functions for all the events that may happen during the whole operation cycle of the power plant. Events which are postulated for NPP by designers are called DBAs. An example of the DBA is the loss of coolant accident (LOCA) or violation of electricity supply. Certainly, the passive safety systems (as well as the active systems) should always be ready for these normal and abnormal events. [18]

## 2.1 List of passive safety systems applicable in the reactors on NPPs

Cooling the fuel and taking away the reactor decay heat appears to be the main function for passive systems in order to cope the DBAs effectively. Scientists from different countries in development of nuclear energy programs have come to different decisions about the way to ensure safety of NPP by using the passive systems. Thus, the number of possible approaches used in new reactor concepts is vast.

Below the main facilities for core decay heat removal of NPP is listed and briefly described.

### 2.1.1 Pre-pressurized core flooding tank

The pre-pressurized core flooding tank is used in existing NPPs. It is a fragment of the emergency core cooling system. The concept is presented in figure 2. Tanks are separated from the primary side by isolation valves and are filled with borated water (3/4 of volume) and pressurized nitrogen or an inert gas (1/4 of volume). During the normal regimes of operation the isolation valves are locked due to the pressure dissimilarity of the gas in the pre-pressurized core flooding tank and the reactor coolant system. In case of LOCA the primary side pressure decreases beneath the pressure of the tank, it causes the isolation valves opening, and discharge the water of the tank with boric acid into the RPV. [14] The tanks are necessary in case of large break LOCAs since it is obligatory to have a higher makeup flow to refill the downcomer and RPV lower plenum initially after reactor coolant system (RCS) blowdown [15].

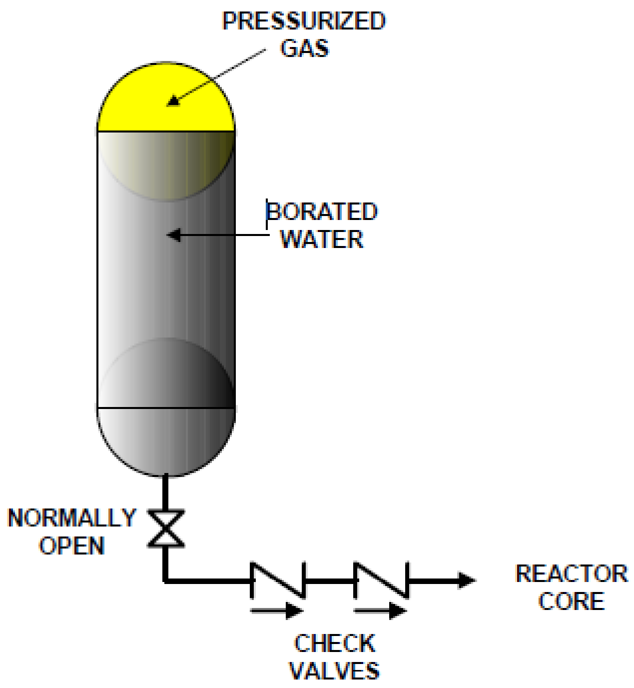


**Figure 2**. Pre-pressurized core flooding tank [14].

### 2.1.2. Elevated tank natural circulation loops (core make-up tanks)

In this system, which is presented in figure 3, the core make-up tank (CMT) is attached to the RPV both at the bottom and at the top. The tank is filled with borated water. In the top connection line the isolation valve is normally open and the line is used for monitoring full system pressure. In the bottom connection line an isolation valve is normally closed and, in case of an accident, the bottom isolation valve will open to allow cold borated water flowing to the reactor's core at system pressure. [14]

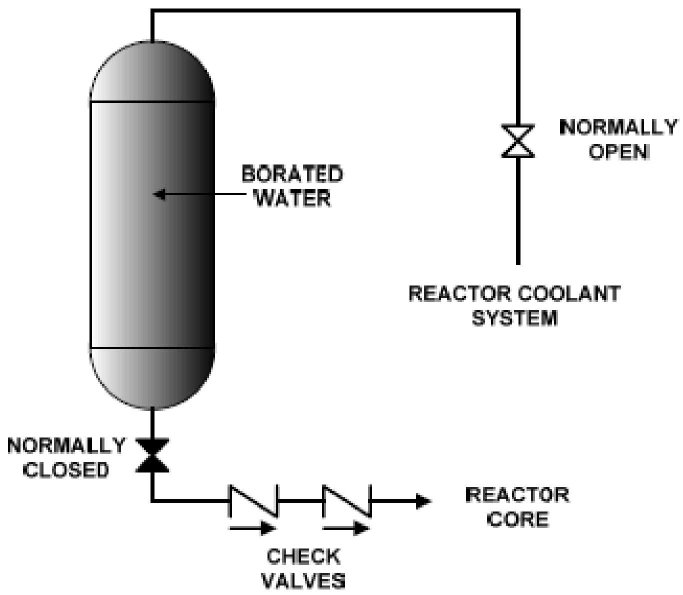


**Figure 3**. Elevated tank natural circulation loops [14].

### 2.1.3. Elevated gravity drain tanks

This system use gravity power to flood the core in case of low pressure conditions. The main component is the elevated cold borated water tank, which might be fairly large even for fulfillment of the entire reactor cavity [19]. To start a process the system needs opening of the isolation valve and exceeding the system pressure by the driving head of the water. In addition, the cracking pressure of the isolation valves should be exceeded [14]. The scheme of the system is presented in figure 4.

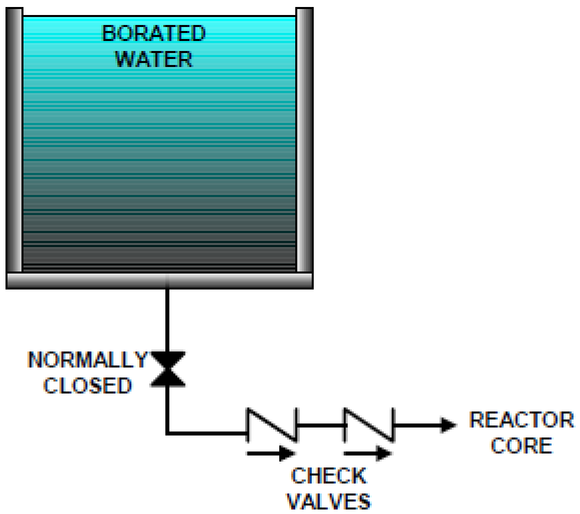


**Figure 4**. Elevated gravity drain tank. [14]

### 2.1.4. Passively cooled steam generator natural circulation

The objective of this system is removing decay heat from the core. The decay heat is taken away through the SG by condensing secondary side steam of the SG inside the heat exchanger located inside the water tank or in an open air system (amount of surrounding air is assumed unlimited) [14]. The scheme of the system for core decay heat removal is presented in figure 5.

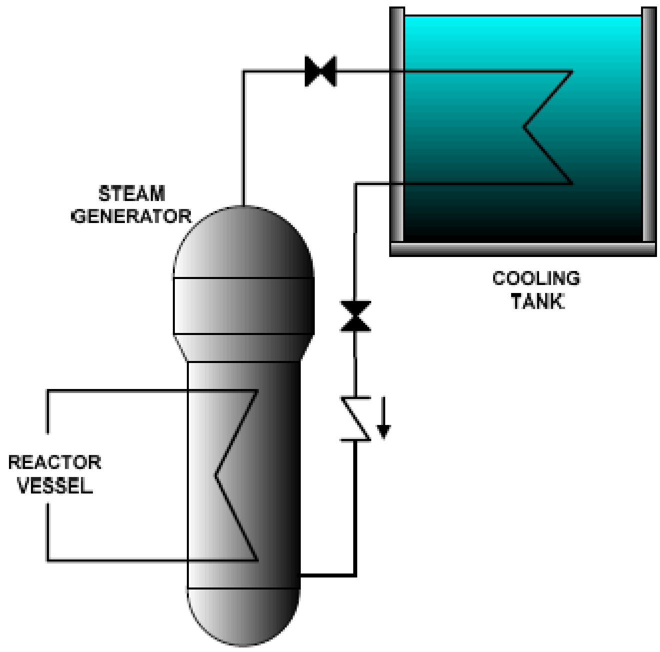


**Figure 5**. Core decay heat removal system using a passively water-cooled steam generator. [14]

### 2.1.5. Passive residual heat removal heat exchangers

Passive residual heat removal (PRHR) heat exchangers can take away decay heat from the core in the event of unavailability of feedwater systems or the SG heat removal. PRHR heat exchangers are used mainly in PWR designs. The system provides a long time removal of the heat from the reactor facility during the accidents involving full and partly loss of electricity at the NPP. The system consists of cooling tank with PRHR heat exchanger and pipes connecting primary system to the PRHR heat exchanger. The water from the reactor vessel flows through to PRHR heat exchanger and conducts its heat to the cooling tank. [19] Water flow is activated by the bottom check valve of the PRHR heat exchanger opening [14]. The scheme of the structure is presented in figure 6.

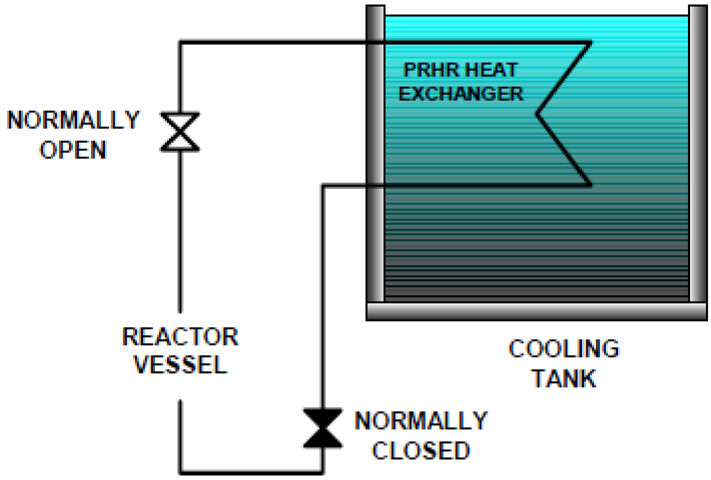


**Figure 6**. Core decay heat removal system using a water-cooled passive residual heat removal (PRHR) heat exchanger loop. [14]

### 2.1.6 Sump natural circulation

This approach provides the cooling of the core in LOCA event. The concept is presented in figure 7. The reactor cavity and other spaces in the lower part of containment are used as a reservoir for coolant. Therefore, the water mass from the primary system flows inside the containment sump. Ultimately the RPV is fulfilled with liquid and all check valves are opened. The natural circulation is formed due to the difference of densities of water in the reactor core and in the containment. Water flows up over the sump screen to the RPV and boils. The steam produced in the process flows up and outputs straight into the containment after passing an automatic depressurization system valve (ADS). [14]

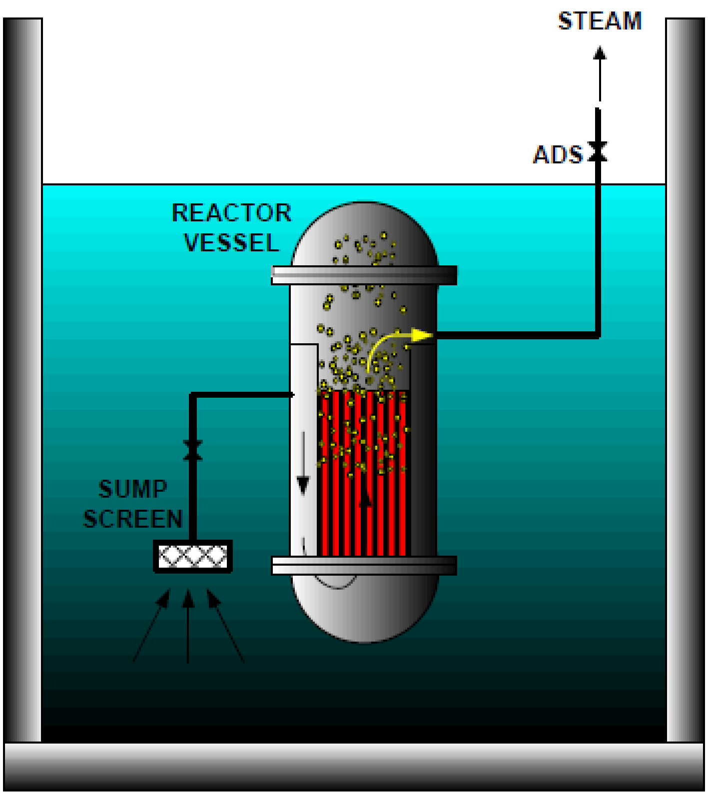


**Figure 7.** Core cooling by sump natural circulation [14]

### 2.1.7 Containment pressure suppression pools

These pools are proven to be effective in boiling water reactor (BWR) designs to prevent the pressure increasing in the containment. The concept of the system is presented on figure 8. In case of LOCA the water from the primary side vaporizes and vapor flows to the drywell through the break [20]. From the drywell zone, the mix of non-condensable gases and steam is forced to flow through the vent lines which are immersed in the water of the suppression pools. In the suppression pool the water condenses the steam and as a result the pressure inside the containment decreases. [14]

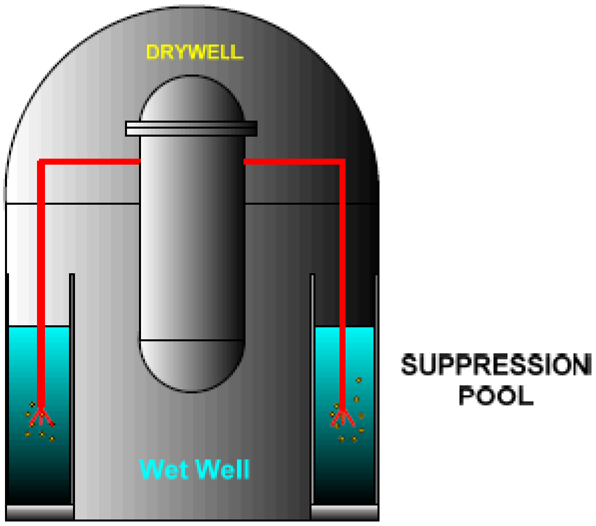


**Figure 8**. Containment pressure reduction after a loss of cooling accident by the steam condensation in the suppression pools. [14]

### 2.1.8 Containment passive heat removal/pressure suppression systems

In this passive safety system the heat sink is represented by an elevated pool. The system condensates the steam inside the containment on the surface of condenser tubes to ensure containment cooling and pressure suppression. This approach has three variations, which are presented in figures 9, 10, 11.

The first variation of the concept is presented in the figure 9. Above the containment there is a water pool attached to the heat exchanger. The water from the pool flows inside the tubes of the heat exchanger while on the outside of the tubes there is atmosphere of the containment. During the LOCA the hot steam condensates on the tube outer wall and the heat of the steam is removed to the water inside the tubes. Due to the incline of the tube and the density difference of the warm and cold water, the warm water inside the tube starts to flow upward and the cold water from the pool starts to flow downwards, forming the natural circulation inside the heat exchanger. [14]

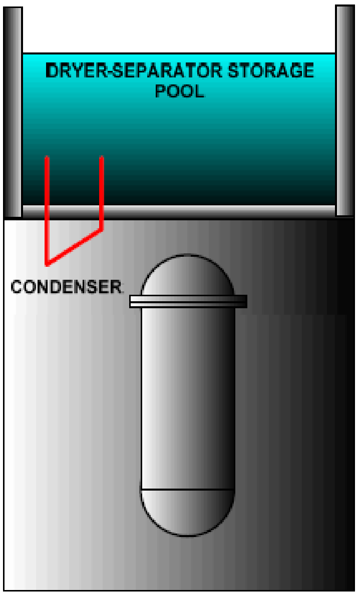


**Figure 9**. Containment pressure reduction and heat removal after a loss of coolant accident by steam condensation on condenser tubes [14].

The second variation of the concept is presented in the figure 10. This concept is very similar to the one in figure 9. This approach uses also natural circulation to perform its function but in this case the natural circulation loop is closed, unlike in the first variation where the natural circulation loop was open. The loop is filled with liquid and it is connected to the water pool and to the air heat exchanger. A difference between densities in the riser and downcomer appears when heat is received from the containment side by air heat exchanger. This heat transfer leads to the natural circulation of working fluid through the closed loop. [14]

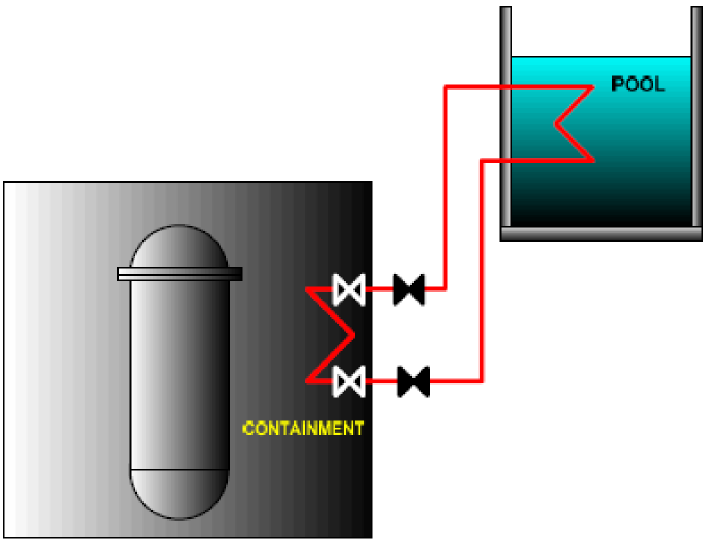


**Figure 10**. Containment pressure reduction and heat removal after a loss of coolant accident by a closed external natural circulation loop [14].

The third variation of the concept is presented in the figure 11. In this concept the natural circulation loop is open and the working fluid in now water-steam mixture. In case of LOCA the steam situated inside the containment is flowing to the heat exchanger located inside the water pool. The steam is condensed inside the tubes when the heat of the steam is conducted through the tube wall to the cold water of the pool. The resulting condensate is flowing back to the containment in the wetwell through the downcomer. The driving force of this system may be lower than in the variations one and two. [14]

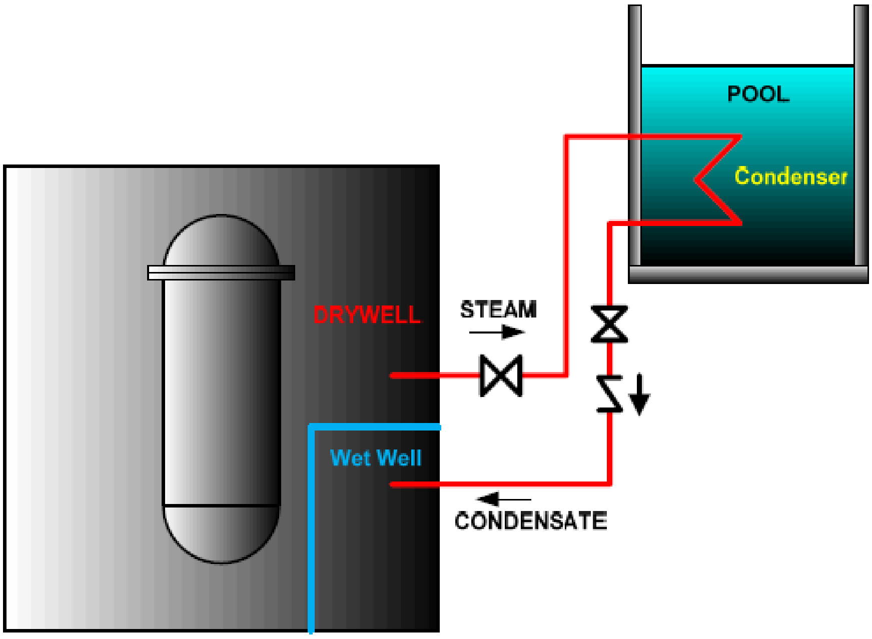


**Figure 11**. Containment pressure reduction and heat removal after loss of coolant accident by an external steam condenser heat exchanger [14].

### 2.1.9 Passive containment spray system

The passive containment spray system implements natural draft air cooled containment. The concept is presented in figure 12. In case of LOCA, the steam inside the containment will condense during interaction with the containment's inside surface. The heat will transfer from the steam to the open air through the wall. The warmed air will flow upwards out of the cooling annulus. In the spray system the water from the pool at the top of the containment is sprayed on the steel containment to provide cooling of the containment. [14]

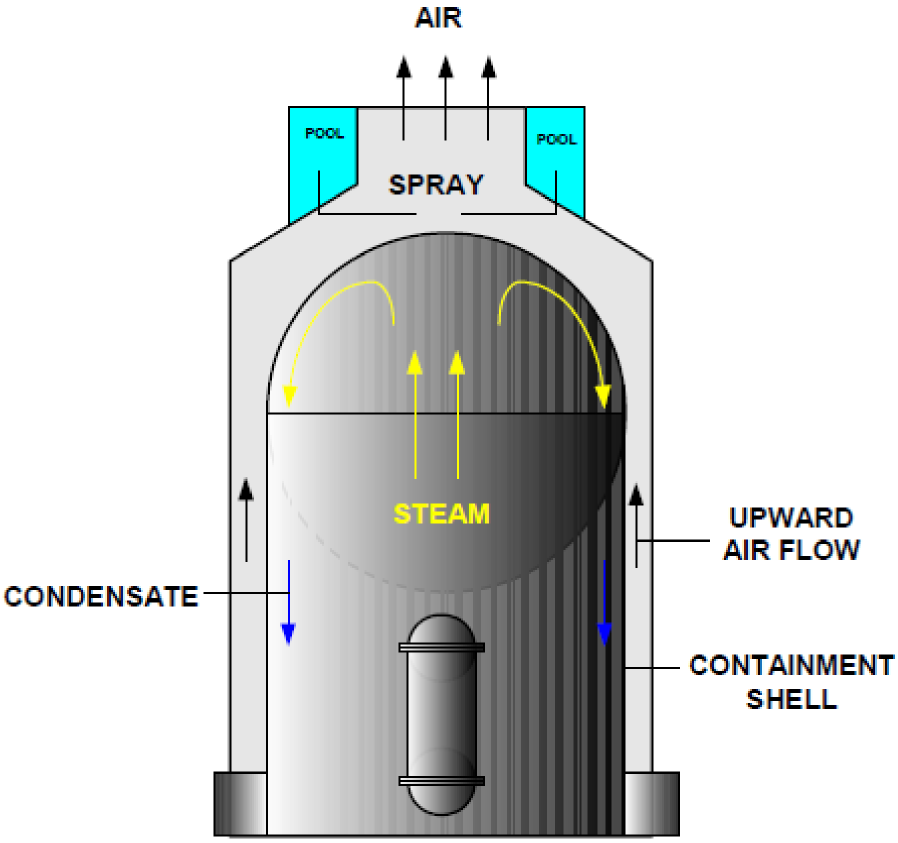


**Figure 12**. Containment pressure reduction and heat removal systems: a passive containment spray and natural draft air [14].

### 2.2 Balance between passive and active systems' application

As it is said before, the passive systems have multiple advantages compared to active systems (see chapter 1) and there is a solid technical background information and operational experience as a foundation for using passive systems in new reactor concepts. Nevertheless, the passive way of operating systems does not mean that reliability of such system would be higher with regard to fulfillment of the designated safety function [8].Therefore there is a need in providing of a certain balance between active systems and new passive analogues. The final design decision should be defined with respect to two requirements [21]:

- To improve safety and ecological acceptability of nuclear power;
- To keep nuclear power competitive with new power technologies, especially renewable ones.

The design of active/passive decisions, which are applied to the NPP, should be chosen with regard to the functions given to the system. Particularly, if the system on the NPP will have an important role in restraining of severe aftermath in a potentially contaminated area, then it should be as independent as possible. This should be done because of the possible difficulties connected with human access to contaminated areas for the long time. [21]

These aspects are being taken into account by nuclear power plant designers, and as a result, the passive systems are used for decay heat removal in NPPs all over the world [9]. In addition, such systems are implemented in modern reactor concepts, and one example is given below.

# 3. CONCEPT OF AES-2006

The AES-2006 is a project of the Russian generation III+ NPP with improved technical and economic indices. One of the main features of AES-2006 is to combine additional passive safety systems and traditional active systems [22]. The scheme of the AES-2006 concept is shown in figure 13.

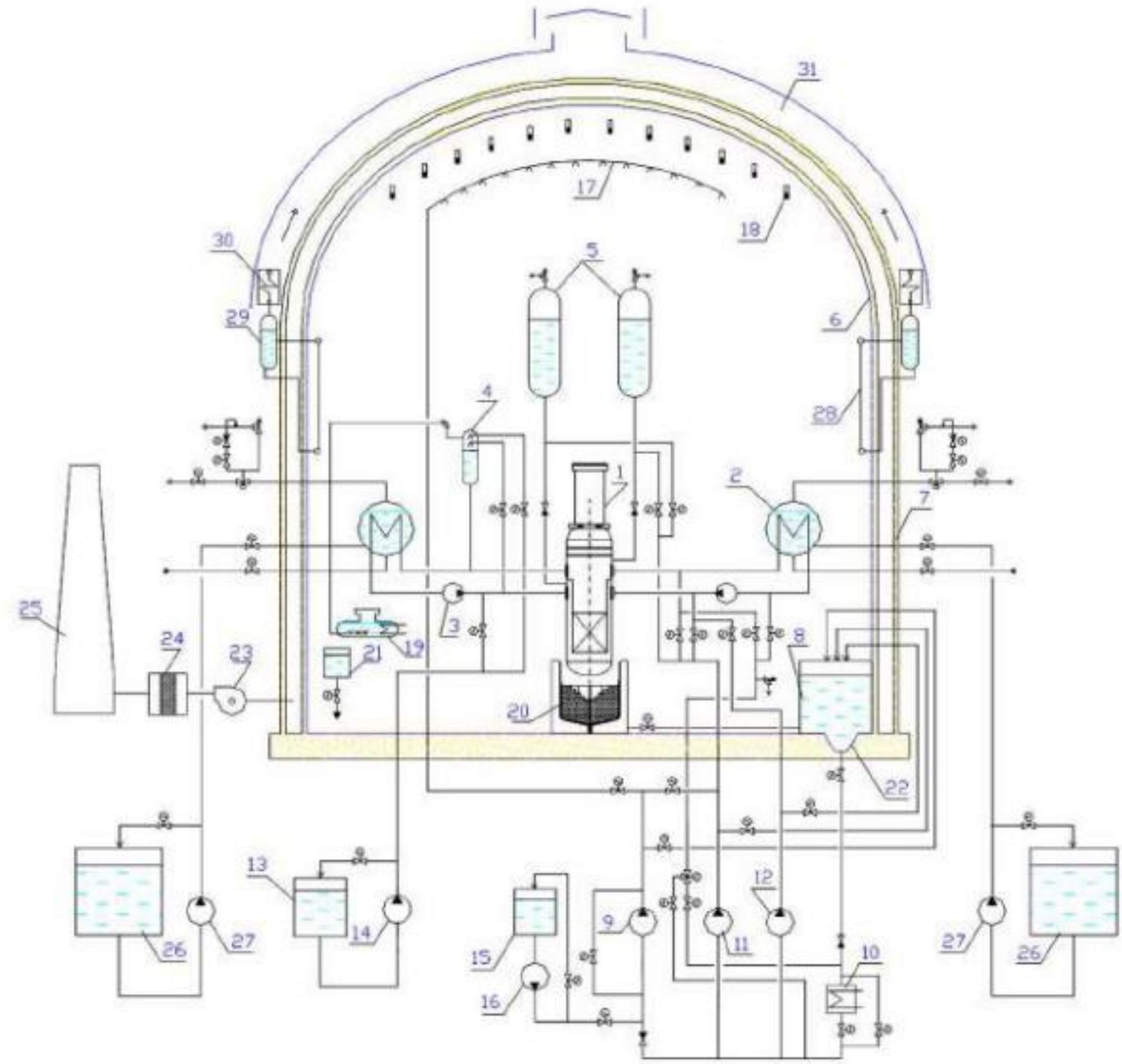


**Figure 13.** Scheme of AES-2006 concept [22]. Numbers in the figure means: 1 - Reactor, 2 - Steam generator,, 3 - Main circulation pump, 4 - Pressurizer, 5 - Emergency core cooling systems tanks, 6 - Protective shell, 7 - Outer protective shell, 8 - Reserve tank for low-concentration borated water, 9 - Pump of sprinkler system, 10 - Heat exchangers, 11 - Emergency low pressure injection pump, 12 - Emergency high-pressure injection pump, 13 - Reserve tank for high-concentration borated water, 14 - Emergency boron injection pump, 15 - Tank for chemicals supplying, 16 - Pump for chemicals supplying, 17 - Sprinkler header, 18 - Passive hydrogen recombiner, 19 - Bubbler, 20 - Melt localization device, 21 - Emergency reserve alkali tank, 22 - Protecting shell's pit, 23 -

Ventilation setting for emergency vacuum creating in the annular space, 24 - Filter, 25 - Ventilation tube, 26 - Reserve tank of demineralized water, 27 - Emergency feed water pump, 28 – Condenser of passive safety system for heat decay removal (SPOT system), 29 - SPOT tank, 30 - SPOT air exchanger, 31 - SPOT air pipe.

## 3.1 General information of the AES-2006 reactor

The reactor of the AES-2006 NPP is WWER-1200 with electric power of 1150 MW (and the possibility of increasing to 1,200 MW). The planned level of installed capacity utilization (load factor) is 92% and the time between refueling up to 24 months. The WWER-1200 (V-491) design was established by “Atomenergoproekt” Company, (St.Petersburg). [23] The structure of WWER-1200 RPV is shown in figure 14. Table 1 contains the specification of WWER-1200 reactor.

**Table 1.** Specification of WWER-1200 reactor. [24]

| | |
|---|---|
| Nominal thermal power of the reactor, MW | 3200 |
| Loops, pcs. | 4 |
| Primary pressure, MPa | 16,2 |
| Secondary pressure, MPa | 7,0 |
| Reactor inlet temperature of coolant, °C: | 298,2 |
| Reactor outlet temperature of coolant, °C: | 328,9 |
| Reactor coolant flow rate, $m^3/h$ | 86 000 |
| Fuel assemblies, pcs. | 163 |
| Reactor control rods, pcs. | 121 |
| Steam capacity, t/h | 4 x 1602 |

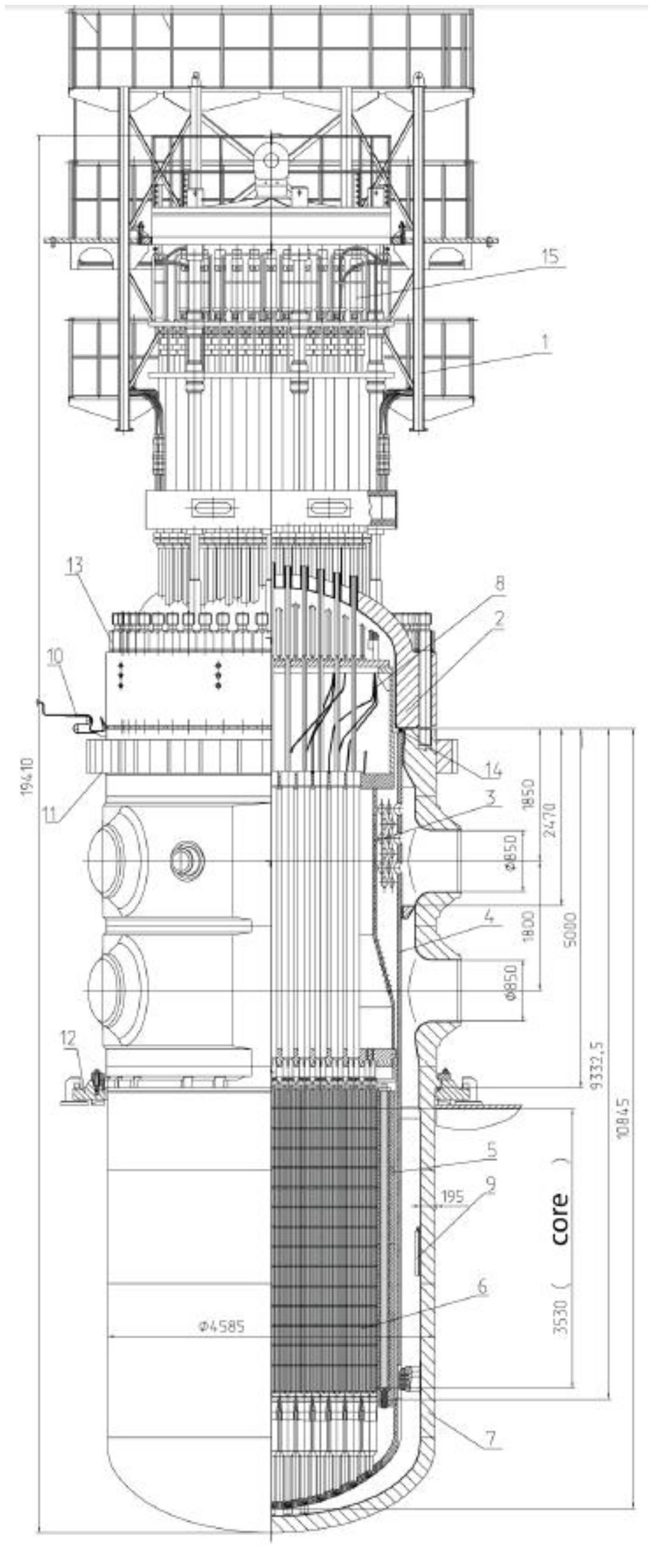


**Figure 14**. Scheme of WWER-1200 reactor [24]. Numbers in the figure means: 1 - Wiring unit; 2 - Upper unit; 3 - Protective tube unit; 4 - Core barrel; 5 - Core baffle; 6 - Core; 7 - RPV; 8 - In-core instrumentation detectors; 9 - Surveillance specimens; 10 - Main joint leak monitoring device; 11 - Thrust ring; 12 - Support ring; 13 - Reactor main joint sealing components; 14 - Pressing device; 15 - Rod Cluster Control Assembly drive

The list of specific structural features which are applied to the AES-2006 generation III+ design include [25]:

- Horizontal SGs with larger water reserves and developed conditions of the natural circulation on primary side in comparison with the vertical SGs;
- Emergency core cooling systems constructed on the active and passive principles;
- Improved reliability I&C containing functions of self-testing;
- Passive components, isolation, restraints and discharge devices.

## 3.2 Safety systems of AES-2006

The AES-2006 reactor includes both the active and passive safety systems in balanced combination, which is a unique feature of this NPP. Such an approach guarantees that fundamental safety functions, like removing of decay heat, will be executed in all situations, including complete loss of electric supply and simultaneous loss of coolant. [22]

The list of safety systems of AES-2006 is presented below [24]:

1. Low and high pressure safety injection system
2. Overpressure protection system for primary and secondary circuit
3. Emergency gas removal system
4. Emergency boron injection system
5. Emergency feedwater system
6. Borated water storage system
7. Core catcher (corium localization system)
8. Containment hydrogen removal system
9. System of passive heat removal from containment (SPOT ZO)
10. System of hydroaccumulators of the first stage and second stage (GE-1 and GE-2)
11. System of passive heat removal through SGs (SPOT PG).

### 3.2.1 Low and high pressure safety injection system

The function of the low pressure system consists in bringing a boric acid solution to the coolant system of the reactor in case of LOCA when the pressure of the coolant system drops beneath the working parameter of the system (below 79 bars) [25].

The purpose of the high pressure system consists in bringing a boric acid solution to the RPV in case of LOCA at coolant system pressure beneath the established limits. Moreover, part of tubes and system equipment prevents release of radioactive components beyond the containment. [25]

### 3.2.2 Overpressure protection system for primary and secondary circuit

The overpressure protection system for primary side is projected to avert the excessive overpressure in the primary side in case of DBAs and BDBAs. It is performed by the pilot operated relief valve of a pressurizer to release the steam from the pressurizer to the relief tank.

The overpressure protection system for the secondary circuit of this system is planned to prevent overpressure in the secondary side of the SGs and the main steam lines over admissible parameters. [26]

### 3.2.3 Emergency gas removal system

This system is proposed to take away the mix of non-condensable gases and steam from the primary circuit and reduce the primary pressure in conjunction with the pilot-operated safety valve of a pressurizer to reduce the effects of BDBAs and DBAs [25].

### 3.2.4 Emergency boron injection system

This active system is proposed to inject the borated water into the pressurizer. It is intended to carry out the tasks listed below [25]:

- Injection of boric acid into the pressurizer in case of an accident with leak of water from primary side to secondary side;
- Providing of concentrated boric acid solution (40g $H_3BO_3$/kg $H_2O$) injection into the primary circuit for fast transition to subcritical condition;
- Compensation of reduction of primary coolant volume to provide safe reactor shutdown after bringing it into subcritical condition.

### 3.2.5 Emergency feedwater system

This system is proposed to supply feedwater to the SGs under DBAs, when it is unmanageable to supply feedwater from other sources. This system is meant to function in case of accidents connected with the water level drop in SGs and necessity of emergency cooldown or maintenance of the reactor in a hot reserve. [25]

### 3.2.6 Borated water storage system

This system is projected for storing of water volumes with high (40g $H_3BO_3$/kg $H_2O$) and low (16g $H_3BO_3$/kg $H_2O$) concentration of boric acid which is used in different operation regimes of NPP. [26]

### 3.2.7 Core catcher (corium localization system)

Core catcher is designed to manage with BDBAs at the off-vessel stage. The view of the corium localization system and its location under the core is presented in figure 15. The system carries out placement, intake and cooldown of the molten core constituents, internal parts of the reactor, and RPV until full crystallization. [25]

Corium localization system performs various functions [27]:

- Protects the cavity of reactor against thermal and mechanical impact of corium;
- Takes in and stores both solid and liquid corium components;
- Provides corium retention;
- Ensures formation of optimal structure and properties of the melt pool and transition of corium to the solid state;
- Provides heat sink from corium to cooling water passively without any coolant makeup up to 24 hours;
- Minimizes hydrogen and radionuclide release into containment on ex-vessel stage of a scenario with core melting.

The corium localization system in AES-2006 design provides corium confinement and excludes corium discharge outside the containment in any scenario. [27]

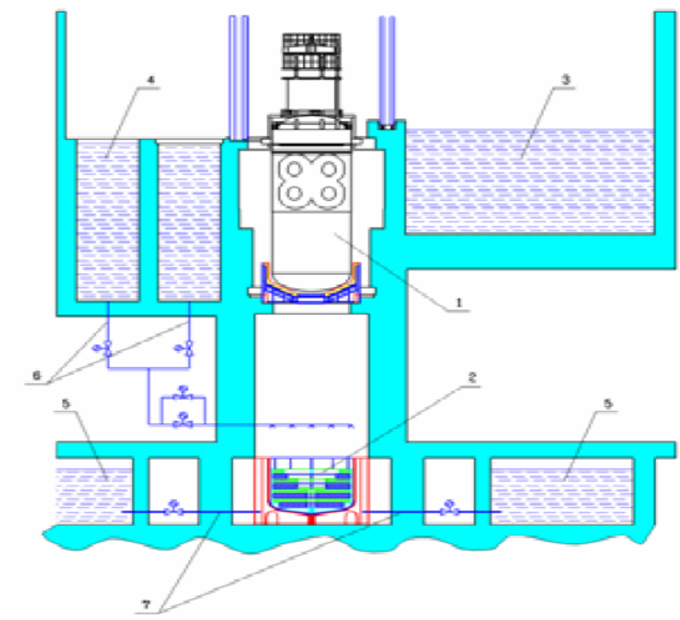


**Figure 15**. Corium localization system, its location under the pressure vessel, and supplying elements on the NPP [28]. Numbers in the figure means: 1 – RPV; 2 – Corium localization system; 3 – Fuel pools; 4 – Inspection vault for in-vessel components; 5 – Sump tanks; 6 – Pipeline supplying water onto corium surface; 7 – Pipeline supplying water to core catcher heat exchanger.

### 3.2.8 Containment hydrogen removal system

The structure contains a set of passive autocatalytic hydrogen recombiners. Under the DBAs the hydrogen removal system keeps the hydrogen concentration in the mix of steam, water and air lesser than the limit of the flame propagation. It eliminates a probability of detonation of hydrogen and a progress of rapid combustion in a large space, which are similar to the sizes of the containment. The capacity of the safety system is designed as if 1000 kg of $H_2$ is generated in the containment during 5-7 hours. [28]

### 3.2.9 System of passive heat removal from containment (SPOT ZO)

The passive heat removal system from the containment (SPOT ZO) is used to overcome the DBAs and it is designed for long-term (offline mode – more than 24 hours) heat take away from the containment during accidents. The scheme of SPOT ZO system is shown on figure 16.

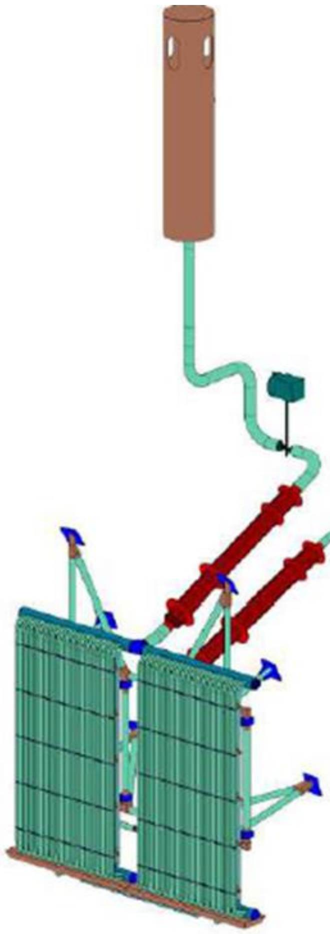

**Figure 16.** Scheme of passive heat removal system from containment (SPOT ZO) [25].

Main functions of the SPOT ZO system are [29]:

- Reducing and fixing the pressure under the containment in the prescribed limits in case of BDBAs, including accidents with core damage;
- Removing of heat from the containment in case of BDBAs to the heat sink, including accidents with severe core damage;
- Ensuring the provision of the sprinkler system to increase the safety of the whole system.

### 3.2.10 System of hydroaccumulators of the first stage and second stage (GE-1 and GE-2)

Hydroaccumulators of the first and the second stage contain boric acid solution to provide extra safety measures in a case of an accident. The accumulator tank of the first stage provides extra supply of boric acid solution to the core at coolant leaks from the primary circuit via discontinuities with a large cross section in case when the pressure of the primary side drops below 59 bars. The accumulator tank of the second stage supplies boric acid solution to the RPV in an accident with of a pressure drop in the primary side beneath 15 bars. [30] The hydroaccumulator system is presented in figure 17.

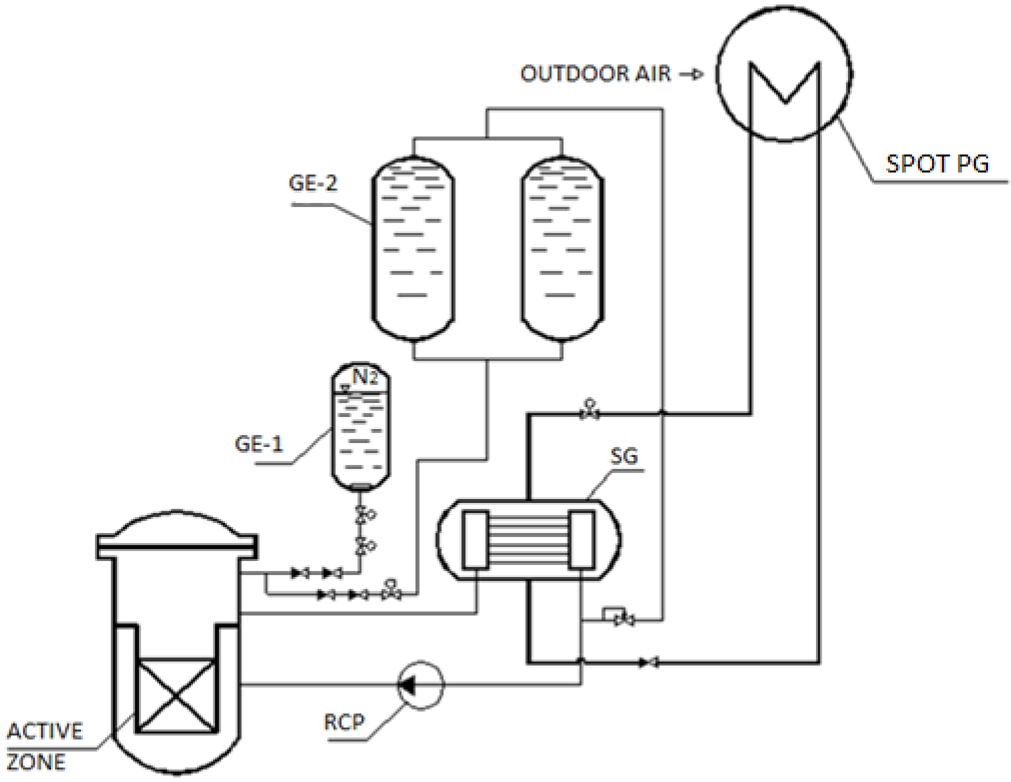


**Figure 17**. Scheme of the hydroaccumulator system and the passive steam generator system for heat removing (SPOT PG) for the active zone of WWER-1200 [31].

### 3.2.11 System of passive heat removal through steam generators (SPOT PG)

The SPOT PG is projected for removing of decay heat from the reactor core to the ultimate heat sink through the secondary side of the SG in case of BDBA [32]. In figure 17 you can see the location of the system relative to other active zone safety systems. Figure 18 presents the main view of the SPOT PG system. Inside the emergency heat removal tanks there are 16 heat exchangers sections for each SG. In figure 19 two of these heat exchanger sections are presented.

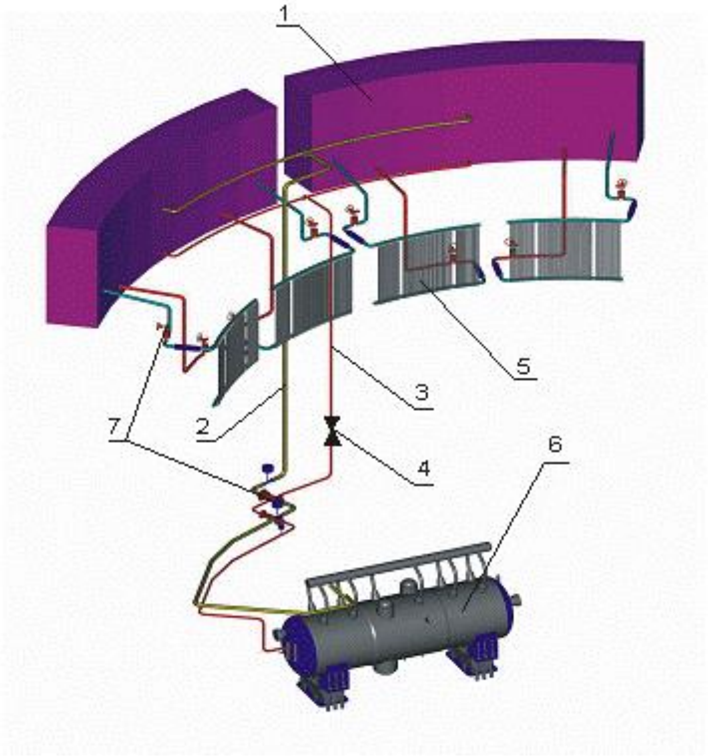


**Figure 18.** Passive heat removal system from the containment (SPOT ZO) and system of passive heat removal through the SG (SPOT PG). [25] 1 – Tank for emergency heat removal; 2 – Riser; 3 - Downcomer; 4 - Valve; 5 - Heat exchangers of SPOT ZO; 6 – Steam generator; 7 - Shut-off valves.

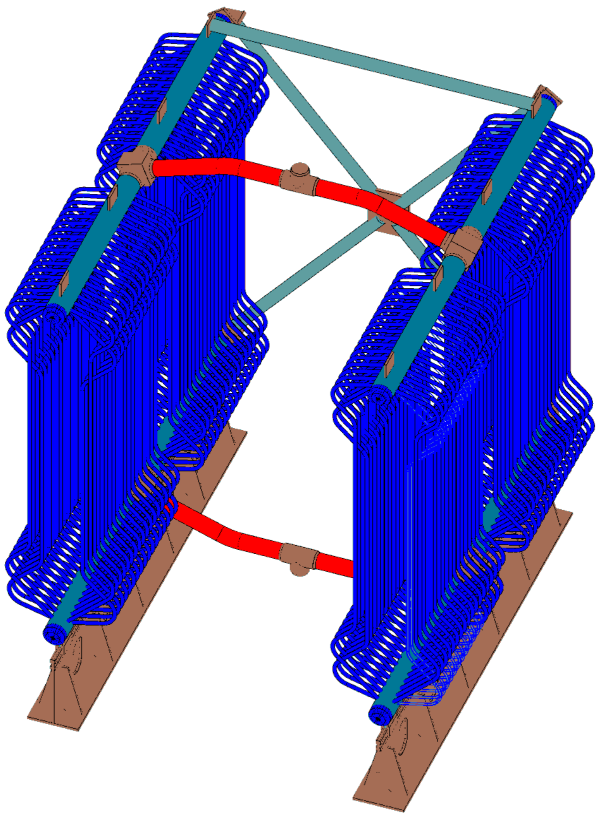

**Figure 19.** Two heat exchanger sections of SPOT PG system. [32]

The SPOT PG system works both as a standalone and in conjunction with other safety systems. The SPOT PG contains four independent natural circulation circuits with 33,3% capacity. Every circuit is connected to its own SG from the secondary side. The geometric characteristics of the SPOT PG system are listed in table 2. Each of four circuit includes the heat exchanger modules, the steam/water condensate pipes connecting the heat exchanger modules to the secondary side of the SG, channels which discharge heated air from the heat exchange modules and air flow regulators. The total height difference of the SPOT PG loop is over 40 meters. [30]

The SPOT PG system is intended for cooling of the active zone, i.e. for long-term take away of the decay heat from the core through the secondary loop to the heat sink in case of following DBAs [32]:

- Failure of all sources of AC electric supply;
- Failure of the entire supply of feedwater to the SGs;
- The leak of the first circuit with a system failure of the system for emergency core cooling;
- Leakage between the primary and secondary sides.

Furthermore, the SPOT PG system provides a reserve for the active safety systems in case of failure at the cooldown of the reactor facility in emergency conditions.

**Table 2**. Geometric characteristics of the SPOT PG channel. [32]

| | |
|---|---|
| The number of heat exchanger units | 16 |
| Number of tubes in the heat exchanger | 140 |
| The surface area of the heat exchanger, $m^2$ | 14.95 |
| The diameter of the heat exchange tubes, mm | 16x2 |
| The difference in height (rising section), m | 41.1 |
| The difference in height (surging section), m | 46.5 |
| The diameter of the main pipe riser, mm | 273x20 |
| The diameter of the main pipeline for drop area, mm | 108x9 |
| The distance between headers of the heat exchanger, m | 1.95 |

# 4. RESEARCH FACILITIES FOR PASSIVE SYSTEMS OF AES-2006 PROJECT

This chapter will review two large-scale Russian research facilities which are used for testing of the AES-2006 passive safety systems. GE2M-PG stand for the WWER-1200 reactor is used for experiments with the SPOT PG system and the containment experimental installation (KMS) facility is used for the analysis of the SPOT ZO system performance.

## 4.1 KMS experimental installation (SPOT ZO tests)

The KMS experimental installation is a model of WWER containment. This research facility is used for describing thermohydraulic processes under the containment in emergency situations at NPPs and for testing the effectiveness of the passive safety systems of NPPs in WWER reactors. In addition, this facility allows testing the safety concerning hydrogen within the NPP containment using helium as hydrogen simulator. [33]

The KMS experimental installation consists of a protective metal cylindrical containment model with a dome, a free space inside the containment, a passive system for taking away the containment heat (SPOT ZO), and a technological supply of air, steam, and helium inside the containment. In table 3 the main parameters of the KMS stand and the AES-2006 are shown.

**Table 3.** Comparing of AES-2006 and KMS experimental installation parameters. [33]

| | **AES-2006** | **KMS** | **Scale** |
|---|---|---|---|
| The height of containment, m | 67 | 20,9 | 1:2,3 |
| The inner diameter of the protective shell, m | 44 | 12 | 1:3,7 |
| Total volume $m^3$ | 76700 | 1865 | 1:41 |

The model of the containment is made of carbon steel with a thickness of 25 mm (18 mm at the dome part) and it has a free volume of 1865 $m^3$. The maximum pressure for this containment model is 5 bar and the maximum temperature is 150 °C. The minimum and maximum steam flow rate in the KMS containment is 120 kg/h and 4000 kg/h, respectively. The maximum flow rate of the helium and air supply system is 100 $Nm^3/h$ and 600 $Nm^3/h$, respectively. [33]

Figure 20 presents the inner construction and some measures of the KMS containment. The shaded areas on the right side of figure 20 show the cross-sectional shape of the facility zones from B1 to B15.

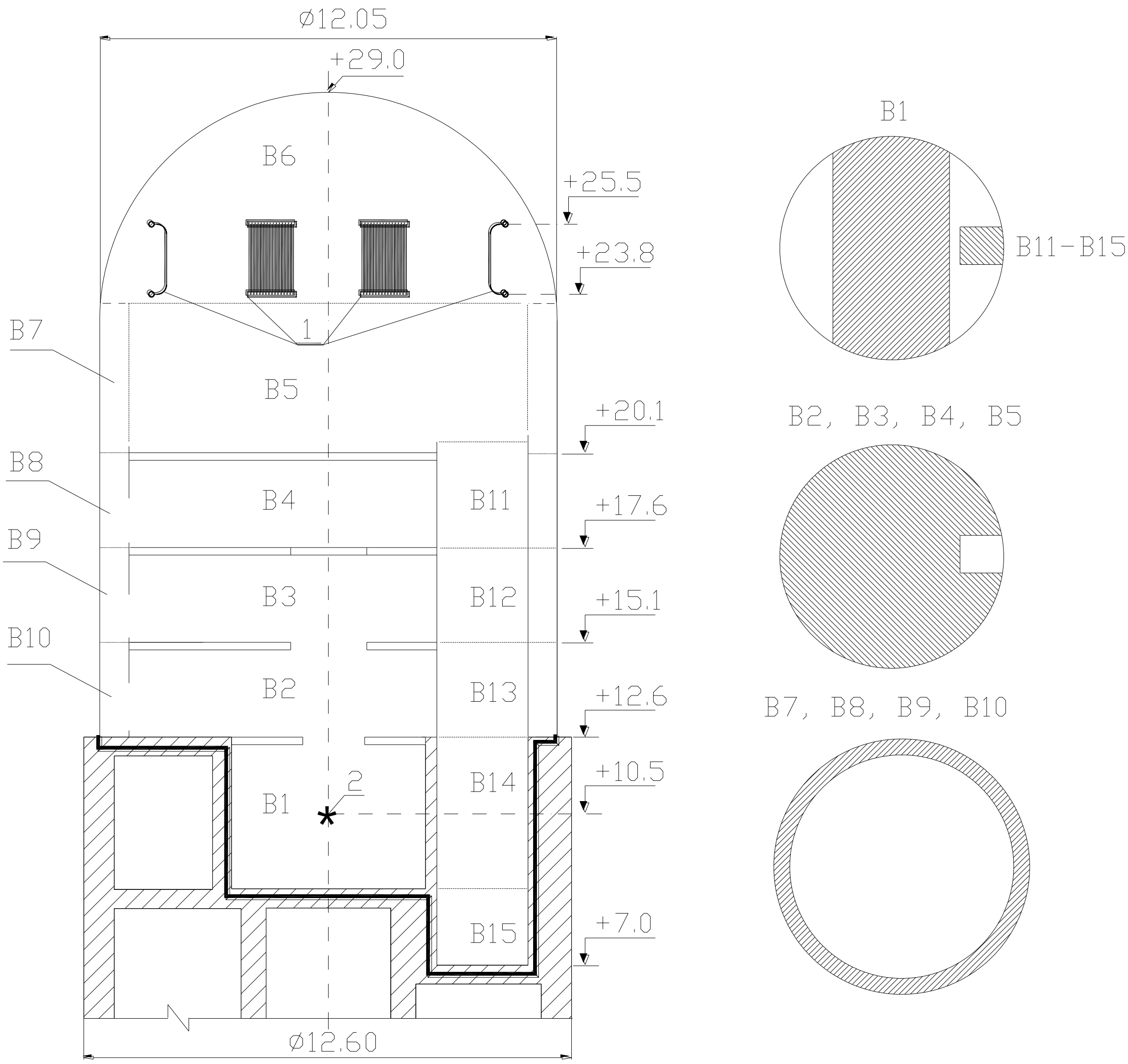


**Figure 20.** Scheme of KMS containment [34]. Numbers in the figure means: B1 – model of emergency pool; B2, B3, B4 – model of reactor's equipment; B5, B6, B7 – space under the dome; B8, B9, B10 – circle gap; B11, B12, B13, B14, B15 – model of fuel pool. 1 – Heat exchangers of SPOT ZO system; 2 – steam and helium injection.

The model of the SPOT ZO system consists of 8 heat exchangers-condensers which are arranged in pairs near the spherical part of containment. The model of heat exchanger-condenser corresponds to the real heat exchanger-condenser of AES-2006 NPP. In figure 21 is presented the picture if the heat exchanger-condenser of the KMS experimental installation. Table 4 contains a comparison of the properties of the heat exchanger of the SPOT ZO in the KMS experimental installation and in the real NPP. [34]

**Table 4.** Comparative characteristics of the heat exchangers-condenser of the SPOT ZO system in the KMS research facility and in the AES-2006 power plant. [34]

| | AES-2006 | KMS | Scale |
|---|---|---|---|
| Number of heat exchanger-condenser sections in the SPOT ZO system, pcs. | 16 | 8 | 1:2 |
| The outer area of the heat exchangers-condenser, $m^2$ | 1200 | ~ 30 | ~ 1:40 |
| Derived capacity, MW | ~ 30 | ~ 0,75 | 1:40 |
| The diameter of the tubes of the heat exchanger-condenser, mm | 38×3 | 38×3 | 1:1 |
| The height of the heat exchanger-condenser tubes (active length, without end bent), m | 4,66 | 1,508 | 1:3,1 |
| The number of tubes in the heat exchanger-condenser unit. | 132 | 20 | 1:6,6 |

The heat exchanger-condenser in the KMS facility is an open-frame heat exchanger, consisting of vertical straight tubes with diameter of 38×3 mm. The tubes are connected on the top and on the bottom by collectors. Cooling water is provided to the lower collector and steam is removed from the upper collector. In addition, there are air vents at the top of the heat exchangers-condensers to remove unwanted air from the system.

**Figure 21.** General view of the model of the heat exchanger-condenser in the KMS facility [34**].**

The whole technological scheme of the KMS research facility is presented in figure 22. Eight heat exchanger-condensers of the KMS facility are situated inside the containment (numbers from 9 to 16 in figure 22). The evaporator tank (or heat sink) is positioned outside and above the containment (number 30). The circulating pump (number 24) is positioned in the downcomer of the SPOT ZO loop.

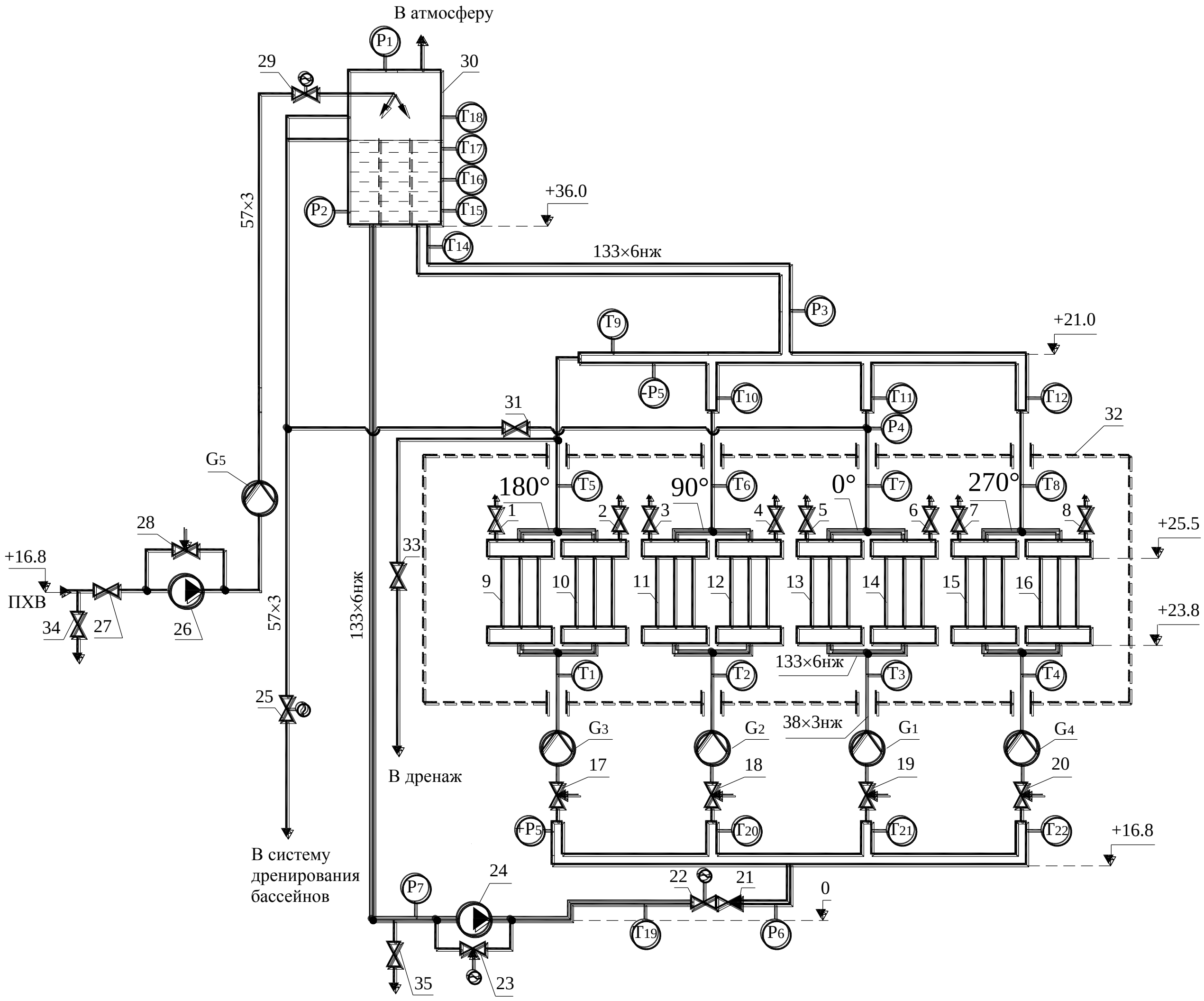


**Figure 22.** Technological scheme of KMS research facility [34]. Numbers in the figure means: from 1 to 8 - vent valves; from 9 to 16 - heat exchanger-condensers; 17, 18, 19, 20, 28 - valves; 21 - non-return valve; 22, 25, 29 - valve with remote control; 23 - control valve; 24, 26 - pumps; 27, 31, 33, 34, 35 - valves; 30 - tank-evaporator; 32 - protective shell (containment); G1 ... G5 - flowmeters; from P1 to P7 - pressure measurements; from T1 to T22 – temperature measurements.

### 4.2 GE2M-PG stand for WWER-1200 reactor (SPOT PG tests)

The large-scale thermal-hydraulic test stand GE2M-PG is designed for tests of the WWER SG in abnormal condensing mode. In addition, it is possible to carry out researches connected with processes of condensation of steam inside the SG in case of non-condensable gases presence. [35]

The stand includes:

- Storage tank with a system of steam;
- Model of SG used on AES-2006, 1:46 scale model;
- Water-cooled SPOT PG heat exchanger-imitator.

The GE2M-PG stand parameters correspond to the project of AES-2006; elevation equipment placement is at the same levels. Equipment and pipelines are insulated to reduce heat loss. Perspective view of the GE2M-PG stand is presented in figure 23.

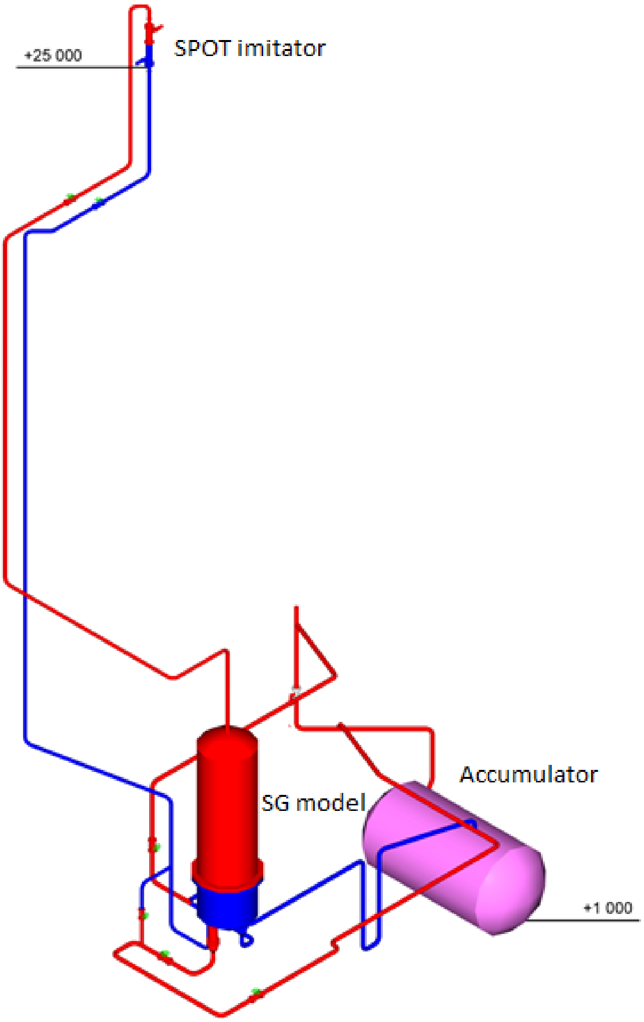


**Figure 23.** Perspective view of GE2M-PG stand [36].

The model of SG is consists of a heat exchanger with two vertical collectors with a diameter of 219 mm (both on "hot" and "cold" side) connected with coiled tube bundle. The coiled tubes are made with a gradient from the center towards the two vertical collectors with the altitude difference of 20 mm to ensure draining of condensate from the tube bundle. The bundle is made up of 248 coiled horizontal tubes in 62 rows with a constant pitch of 36.5 mm for the height of the reservoir. Every row contains 4 pipes with diameter 16x1,5 mm and a length of 10.2 m. Tubes are made of stainless steel. [36] The characteristics of the GE2M-PG stand are introduced in table 4.

The properties of the tubes of the heat exchanger corresponds to the properties of full-scale SG tubes. The surface area of the tube bundle is 48 times smaller than that of the full-scale SG heat exchange tubes. The view of SG model is presented in figure 24.

**Table 5.** The main parameters of the GE2M-PG stand. [36]

| **Name** | **Value** |
|---|---|
| Working fluid | Water, steam |
| Max. pressure, MPa | 1,6 |
| Max. temperature $^0$C | 200 |
| The main equipment of the stand | |
| Model of steam generator | |
| Scale | 1:46 |
| Max. power, MW | 1,0 |
| Number of tubes (rows) | 248 (62) |
| Diameter of the pipe, mm | 16x1,5 |
| Length of the pipe, m | 10,19 |
| Vertical pitch pipe, mm | 36,5 |
| Tube bundle material | Stainless steel |
| SPOT imitator | |
| Max. power, kW | 800 |
| Coolant | Technical water |
| Storage tank | |
| volume, m$^3$ | 16 |

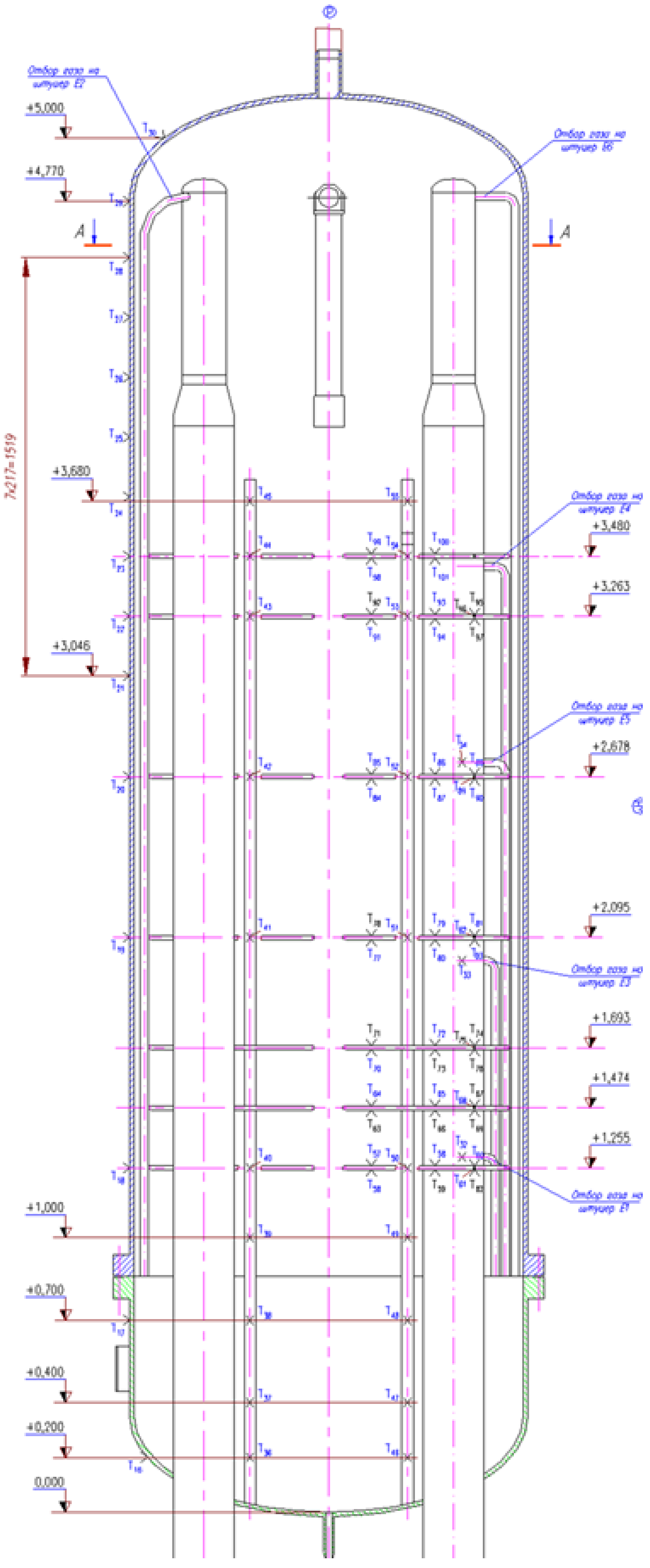

**Figure 24.** Model of steam generator on GE2M-PG stand [36].

# 5. ANALYSIS OF PASSIVE SAFETY SYSTEMS' FAILURE MODES ON NUCLEAR POWER PLANTS

The passive safety systems of NPPs considerably increase the performance reliability of protective functions. To estimate the effectiveness and the adequacy of the measures taken to ensure the reliability, a qualitative and quantitative analysis of the system should be taken. These analyses allow to determine the consequences of the failure of the individual elements or the whole system and to evaluate the reliability index of the whole system. [37]

The purpose of these analysis is to make recommendations on the following issues:

- preventing and reducing the frequency of failures of the elements and the systems;
- estimating the rational multiplicity of redundant elements and the channels of the system to meet the deterministic (single failure principle) and probabilistic reliability criteria;
- eliminating various kinds of dependency between the elements and the potentially possible common cause failures;
- preventing or reducing the chance of the component and system failure owing to human error;
- regulating the periodicity of the system tests and checking of the status of all the elements in the system during operation;
- estimating the permissible duration of the working channel's operation after detecting the malfunction;
- estimating the required level of reliability.

The choice of the method to achieve a desired level of security should be based on system's reliability analysis as well as on the importance of influencing factors and the sensitivity of reliability parameters to change of these factors. In addition, the optimization of economic performance is expedient. [37]

## 5.1 Qualitative & quantitative analysis of the system's reliability

The qualitative analysis of the safety system comprises the following main stages [38]:

- definition of the boundaries, the composition, the functions, and the failure criterion algorithm of the system
- classification and analysis of the elements of the system
- identification of the possible common cause failures

- analysis of the impact of possible human error in the management, maintenance, test, etc. to the state of the system
- determination of the effects of component failures and human errors
- analysis of the structure of the system to identify weak links.

To determine the system's reliability the classification of the elements should be done. The classification process for the elements includes [6]:

- operating principle determination;
- duration of the emergency operation;
- types and causes of failures;
- method of control during reactor operation;
- maintainability;
- impact of periodic monitoring on the performance.

By the principle of action the active and the passive elements are distinguished. The passive components with no moving parts in estimations are often regarded as absolutely reliable. Nevertheless, the failure of these passive components might cause serious accidents.

The principles of the qualitative analysis say that the reliability of any object, including safety systems and individual safety devices, represents a complex set of properties that includes: reliability, durability, maintainability, and retentivity.

The overall analysis of passive safety systems is more focused on reliability and maintainability. These attributes are responsible for system's abilities to continuously maintain a usable state and to restorate through maintenance and repairs, which are among of the most important ones. [39]

## 5.2 Reliability of passive safety systems

In the beginning of the analysis of the passive safety system it can be assumed that the reliability assessment can be estimated by using the probability for having a failure of the function performance of the system. The analysis is held by comparison of the distribution of the values of the expected parameters to the acceptable range. In addition, it involves the identification and quantification of uncertainty in predicting the physical characteristics of the phenomena or dependencies inside the chosen system. An adequate effort should be applied for disquisition of the behaviour of safety systems to integrate those uncertainties. [8]

If we turn to the theory, the passive safety system ought to have a higher level of reliability than an active system. The reasons are comprehensible: it does not need input energy from external sources for performing its function. Besides, the passive systems relies on specific inherent properties of its components, so as on natural physical laws. [9]

In sum, the passive systems' reliability is influenced by [40]:

- environmental phenomena that may affect expected results of operation,
- deviation of physical phenomena from the expected results,
- reliability of every single component of the system.

### 5.3 Failure modes and uncertainties of natural circulation phenomena

The natural circulation is a complicated phenomenon and full of uncertainties. The uncertainties arise from the correlations, which are used to define the physical phenomena. Also, the different types of failure modes can affect the capability of the natural circulation heat removing. These failure modes can be identified as the following [6]:

- Failure due to abnormal stresses, localized stresses, ageing effects of metal pipes, material defects, defects of welding, corrosion; such kind of failures may cause cooling liquid leak, flow rate reduction, and hence, lesser heat removal capability
- Cracking due to material defects, welding defects, localized stresses, corrosion: this failure mode results in decreasing of the heat conduction and leads to a lesser heat removal capability
- Deviation from the initial state of the surface characteristics of the pipes (e.g. oxidation) which results in reduction of heat exchange efficiency
- Thermal stratification; leads to decreasing of heat convection
- Non-condensable gases presence: leads to decreasing of heat exchange efficiency
- Friction inside the pipe, which can lead to blockage of valves; this may reduce flow rate reduction and stop the natural circulation.

The environmental effects that can also cause some uncertainty to the natural circulation, e.g. [6]:

- internal and external incidents (temperature, pressure)
- ageing effects (fatigue, corrosion)
- impurities (gases, liquids, solids)
- corrosion products
- fission products

- radiolysis products
- chemistry of the coolant (pH, $O_2$, $H_2$, boron, etc.)
- irradiation effects.

All these factors may be accounted individually or as a boundary event combination leading to the failure of passively performed function: in every case they determine the performance of the systems based on natural circulation. There are many methods to make an evaluation of the natural circulation heat removal ability and to decrease the connected uncertainties, but the most common are [39]:

1. Providing of the stability of the flow through the circuit of the natural circulation
2. Increasing of the heat exchangers' heat transfer coefficient

The implementation of these requirements can be associated with increasing the inside tube diameter for limiting of the pressure losses initiated by the greater mass flow, reduction of wall thickness in heat exchangers, and the use of components with greater ability for heat conduction. [39]

### 5.4 Natural circulation failure probability

To assess the natural circulation failure probability distribution it is needed to calculate the amount of heat which is possible for the system to remove and to determine the uncertainties related with the heat removal value. This uncertainty can change the behavior of the system and make the chosen correlations useless for future calculations. Thus, the variance includes uncertainties in material and geometric properties and in the chemical/physical phenomena. [39]

To avoid the failure of natural circulation, i.e. the unavailability to take away the decay heat out of reactor core, the relevant parameters must fluctuate above and below the predetermined limits. The list of these parameters includes [39]:

- heat flux from reactor
- coolant fluid temperature
- convection heat flux
- temperature difference between fluid and surface of tubes in SG
- mass flow.

Failure probabilities based on the heat removal value and list of uncertainties above can be assessed. After defining these probabilities it is possible to calculate actual heat flux under the various failure modes.

Failure distribution is defined with assumption of independence of all the failure modes and all the uncertainties of each other. In the end of assessment overall natural circulation probability distribution of failure will be defined by connecting the probability distributions of failure for every failure mode and the mean heat flux at which the natural circulation is predicted to fail can be evaluated. [6]

### 5.5 Classification and causes of failure modes for natural circulation

For analysis of failure modes it is needed to take into consideration the information provided in the Committee on the Safety of Nuclear Installations (CSNI) Report. It was developed for the primary cooling systems that are appropriate for modern NPP concepts. [40]

The list of failure modes for natural circulation includes [40]:

1. Behavior in large pools of liquid
2. Influence of the presence of non-condensable gases on condensation heat transfer
3. Condensation processes on the parts of containment
4. Behavior of the emergency systems of the containment
5. Pressure drops and thermo-fluid dynamics and in different geometrical configurations
6. Interactions between liquid and steam
7. Gravity driven cooling and accumulator behavior
8. Stratification of the water temperature
9. Behavior of isolation condensers and emergency heat exchangers
10. Behavior of CMTs

#### 5.5.1. Behavior in large pools of liquid

Both modern and traditional reactor concepts contain large water pools at pressure about 1 bar for different needs. The large water pool can perform a function of heat sink for taking away of reactor’s heat or from the containment by means of phenomenon of natural circulation. In addition, it can be a water source for cooling of the reactor’s core. [14]

The large water pools may have different geometric forms, which makes them difficult to analyse. The main cause of failures in such conditions is complicated temperature distribution inside the pool. The reason for appearing of such a phenomenon is that the heat transfer in large pools may affect only a limited volume of water. Development of 3D convection flows influences the process of heat transfer and leads to temperature stratification. Eventually it may results in an unsafe situation when the liquid at the surface of the large water pool has already saturated while the main part of liquid still has lower temperature. Such a phenomenon is hazardous because it causes a rise of containment pressure. This kind of temperature stratification requires corresponding modeling, because it can influences the operation of the nuclear power plant noticeably. [41]

### 5.5.2. Influence of the presence of non-condensable gases on condensation heat transfer

Condensation is a thermodynamic process, which take place in case when the temperature of steam drops beneath the temperature of saturation. The presence of non-condensable gases (for example $H_2$, $N_2$, He) in the steam can reduce heat and mass transfer during the condensation heavily. This phenomenon also can be applied in industrial applications and thermal engineering. [42]

In NPPs the described phenomena becomes hazardous during the LOCAs when steam from primary system is released in the containment and is mixed with the air. Also, the nitrogen gas which pressurize the water of the emergency core cooling accumulators can affect heat transfer regime inside the SG tubes of NPPs and the performance of CMT after released to the primary system and may cause overheating and failure of the system. [43]

### 5.5.3. Condensation processes on the parts of containment

This thermalhydraulic phenomenon includes mass and heat transfer from the volume under the containment towards the surrounding elements of the NPP. This condensation during a leak in primary coolant system and when the surface of the containment is cooled with application of passive means, i.e. externally. During the LOCA, large mass of steam is released inside the containment. In case of such an accident the steam starts condensing on the walls of the containment. This occurs because the saturated temperature of steam is higher than the wall temperature. Simultaneously the non-condensable gases generate an additional thermal resistance beside the layer of film condensate. This mix inside the containment reduces the effectiveness of the heat transfer, which is hazardous and leads to overheat. [44]

### 5.5.4. Behavior of the emergency systems of the containment

The containment passive system, which uses natural circulation and condensation heat transfer for removing the extra heat out of the containment, may be the cause of failure too. The major purpose of such system is to protect the compactness of the containment under DBA and accidents with fuel damaging and to avoid releasing of radioactive substances to the containment atmosphere during DBAs and BDBAs. During the accident with core damage, non-condensable gases (e.g. hydrogen) might be released inside the containment, which is hazardous. [45]

Thus, to prevent the failure it is needed to take into account thermohydraulic issues like condensation with the presence of non-condensable gases, the influence of these gases to the natural circulation, deterioration of condensation owing to the increase of the non-condensable gases mass, and removing of such impurities from the condenser system. [44]

### 5.5.5. Pressure drops and thermo-fluid dynamics in different geometrical configurations

The amount of line pressure that is permanently lost from the pipe as the working body passes through is named pressure drop. This loss is initiated by flow resistance and direction, changes in density of the fluid and elevation level. The pressure drop influences on the steady state and stability of the passive safety systems based on natural circulation. The total pressure drop for single phase flow is calculated from its components: local losses of pressure owing to unexpected deviations of the direction and area of the flow, distributed pressure loss owing to frictions of the flow, and pressure losses due to acceleration and elevation. [2]

Significant factors, which affect this phenomenon, are:

- Geometry of the channel
- Number of components in the fluid
- Flow pattern
- Nature of the flow
- Direction of the flow

One of the most difficult dynamic factors to take into account is geometry that may prevent the full development flow establishment in different regimes. Another issue to mention is the driving force which is different in the active (pumps) and passive systems (natural circulation). Under some circumstances which are connected with local fluid properties, the pressure loss might dramatically change the development of the flow. [6]

### 5.5.6. Interactions between liquid and steam

Steam–liquid interface is engaged in containment phenomena. The steam-liquid interaction can be observed in case of steam discharge into a suppression pool of BWR where the formation of bubble plumes takes place after the breaking of the bubbles which were originally generated in the suppression pool. Consequently, a complete condensation of working fluid takes place and causes a mixing of substances in the pool. After that the process in the pool can be defined by natural circulation. [7]

The pressure of the whole system is completely dependent from the steam pressure above the water layer in the chamber of suppression. The temperature of the pool surface strongly depends on the effectiveness of the condensation process inside the water pool, and the efficiency of the components mixing inside the water pool. [2]

Therefore, the list of steam–liquid interactions that may be the cause of failure of safety systems consists of three points [2]:

- Direct contact condensation of steam in the pool;
- Break-up and plume-stirring process and mechanisms inducing mixing in the pool,
- Bubble formation and break-up and the subsequent formation of bubble plumes.

### 5.5.7. Gravity driven cooling and accumulator behavior

This safety concept is created by the RPV depressurization to reasonably low pressure in order to empower the core flooding from an elevated pools. Thus, the concept includes large water pools above the core and capacities for depressurization to establish the stable gravity flow from an elevated pool. This flow floods the lower parts of the RPV and causes steam condensation and boiling suppression. [40]

At the beginning of the flow establishing, the flow rate to the RPV is controlled by the pressure difference of the RPV and the water pool, the geometry of pipes, and the state of working body inside them. These factors may lead to failure of the whole system. The flow must be adequate to hold the reactor core covered with liquid to avoid severe accident. [40]

Accumulating air in the primary side of this passive cooling system before steam is totally condensed can deteriorate the steam condensing in the RPV and drywell gas space. [6]

### 5.5.8. Stratification of the water temperature

The passive safety systems with the natural circulation tend to produce large temperature gradients inside working fluid, which may adversely affect the reliability of the system.

The main reasons for the occurrence of stratification is the presence of local phenomena such as core cooling initiated by emergency cooling system or heating initiated by the steam condensation and heat transfer inside heat exchanger. Typical low powers of the natural circulation flow greatly reduces the fluid mixing and leads to thermal stratification in the pool. During the emergency core cooling the development of cold plumes in the down comer tube, which can stratify the working fluid temperature in the cold legs and lower plenum may be revealed. [40]

Other systems where the liquid temperature stratification can appear are CMT tanks and large water pools which have an ability to perform a heat sink function for containment passive cooling systems. For example, when steam enters into the tank it condenses in the cold liquid and could create a plumes that develop a layer at the water surface of the CMT. The layers inside the fluid will have different temperatures that may result in a large temperature gradient in the liquid and severe working conditions for metal elements [6]. Also, in the situation where the tank includes a subcooled water and saturated steam at the same time, the liquid could condense the steam in the tank and the subcooled water layer could stay below a layer of saturated water. The temperature of saturated layer could differ from the subcooled layer dramatically and results in a stratified temperature condition in the fluid. [40]

### 5.5.9. Behavior of isolation condensers and emergency heat exchangers

The decay heat removing from the PWR reactor may be performed by natural circulation with systems like an isolation condenser or an emergency heat exchanger. It is taken away by convective heat transfer phenomenon in the RPV. Generally speaking, the process of heat transfer from the working fluid to the pool through the tubes of emergency heat exchanger is implemented by three main tools [7]:

- Heat conduction through the tube walls,
- Heat transfer via convection at the inner surface of the tubes,
- Nucleate boiling at the outer surface of the tubes.

Failures of this system may be connected with such phenomena as loop flow resistance, natural circulation termination and ineffective single-phase convective heat transfer [2].

The isolation condenser is included for removing core decay heat in advanced BWR projects. This system contains a tube heat exchanger submerged in an elevated water pool and a shell. The decay heat is taken away from BWR by nucleate boiling. During this process produced steam is condensed inside the tubes of isolation condenser. This leads to formation of an area of low pressure in the tubes; it retracts an extra steam mass. Therefore, steam condensation process is the main driving factor for performing safety function. For isolation condenser there are three principles of heat transfer through the tubes into the pool [40]:

- The conduction of the heat from the tubes to the water pool,
- Heat transfer via convection at the outer surface of the tubes,
- Condensation of the steam at the inner surface of the tubes.

In addition, the action of the system might be violated by the non-condensable gases inside the working body and counter current flow limitations for condensate and steam. [46]

### 5.5.10. Behavior of CMTs

Top and bottom of CMTs are attached to the primary loop and RPV. The tanks are filled with cold borated water. [46]

In case of an emergency, the bottom check valve of CMT is unlocked to create a loop of natural circulation in order to allow the flow of cold borated water into the core. The flow is created by means of the difference of the CMT elevation over the core and the difference of densities in the water inside the primary system and inside the CMT. [46]

Since CMT behavior includes the phenomena like natural circulation, gravity, liquid flashing during plant depressurization and thermal stratification, the system appears to be one of the complex ones to research for possible failures [6].

# 6. MODELLING OF SPOT PG SAFETY SYSTEM

In this chapter the thermal-hydraulic system code calculations for the SPOT PG safety system is presented. The primary purpose of these calculations is to test the behavior and understand the performance of the system. In these calculations three parameters are investigated as a function of decay heat power: 1) the pressure of the system, 2) the natural circulation loop mass flow rate, 3) the level of water in the downcomer. Another purpose is to check how efficiently the SPOT PG can take away the decay heat from the AES-2006 reactor primary side. The TRACE (TRAC/RELAP Advanced Computational Engine) code has been selected to carry out the calculations in this work.

## 6.1 Description of modeling tools

### 6.1.1 TRACE

TRACE (TRAC/RELAP Advanced Computational Engine) is a thermal-hydraulic system code which is developed by the U.S. Nuclear Regulatory Commission (NRC).

TRACE code takes a component-based method to modeling. Physical equipment included in a flow loop can be modeled as a component of the code, and nodalized into cells. After that the equations for conduction and fluid are averaged over them. The code is used for modeling of fuel behavior, multidimensional two-phase flow, reactor kinetics, and passing of thermal-hydraulic processes during postulated conditions or accidents. [47]

### 6.1.2 SNAP

SNAP (Symbolic Nuclear Analysis Package) is an interface which is also developed by U.S.NRC. It represents a combination of integrated tools considered to make the process of thermal-hydraulic analyzing even simpler. This interface offers an opportunities to create and edit the input for thermal-hydraulic codes and instruments for interacting with them. The SNAP simplifies and eases the work of developing TRACE models by automatically generating input files. The input file is processed in such a way that makes it much easier to interpret by a model developer. In addition, the SNAP allows creating animation models which helps to present and interpret the results of the calculations. [48]

## 6.2 Description of the TRACE model

The SPOT PG system is described previously in chapter 3. Each of the AES-2006 SGs has one of these systems. The system contains the heat exchanger-condenser immersed in the heat sink tank, the secondary side of the SG, and piping between these two (the riser and the downcomer) (see figure 18). The heat exchanger-condenser for each system consists of 16 heat exchanger sections (see figure 19) and each section has an upper and bottom header connected with 140 heat exchange tubes.

The TRACE model of the SPOT PG involves the SG, the heat-exchanger condenser, the riser and downcomer pipes, and the heat sink water pool. Figure 25 presents the general view of the TRACE model and table 6 presents the geometric information for the different components. The geometric information is obtained from the AES-2006 project data (see chapter 3) and from the research facility description (see chapter 4.2). The parameters of the model are close to the values of the real SPOT PG system. The full graphical representation of model can be seen in Figure 25.

**Table 6.** Dimensions used in the TRACE model.

<table>
<tr><td rowspan="3">Riser</td><td>Height, m</td><td>39,375</td><td>Approximately [38]</td></tr>
<tr><td>Diameter, m</td><td>0,233</td><td>[38]</td></tr>
<tr><td>Length, m</td><td>45,0</td><td>Estimated</td></tr>
<tr><td rowspan="3">Downcomer</td><td>Height, m</td><td>40,672</td><td>Approximately [38]</td></tr>
<tr><td>Diameter, m</td><td>0,09</td><td>[38]</td></tr>
<tr><td>Length, m</td><td>45,191</td><td>[38]</td></tr>
<tr><td rowspan="9">Heat exchanger-condenser of SPOT PG system</td><td>Number of units</td><td>16</td><td>[38]</td></tr>
<tr><td>Number of tubes in one unit</td><td>140</td><td>[38]</td></tr>
<tr><td>The distance between headers of the heat exchanger, m</td><td>1,95</td><td>[38]</td></tr>
<tr><td>Length of the header, m</td><td>2,124</td><td>Estimated</td></tr>
<tr><td>Diameter of the headers, m</td><td>0,08</td><td>Estimated</td></tr>
<tr><td>Surface area of the unit, $m^2$</td><td>14,95</td><td>[38]</td></tr>
<tr><td>Inner and outer diameter of the tube, m</td><td>0,012/0,016</td><td>[38]</td></tr>
<tr><td>Surface area of one tube, $m^2$</td><td>0,107</td><td>Calculated</td></tr>
<tr><td>Length of the tube, m</td><td>2,12</td><td>Calculated</td></tr>
<tr><td rowspan="3">Steam generator</td><td>Diameter, m</td><td>6,18</td><td>51</td></tr>
<tr><td>Length, m</td><td>4,2</td><td>51</td></tr>
<tr><td>Volume, $m^3$</td><td>126</td><td>51</td></tr>
</table>

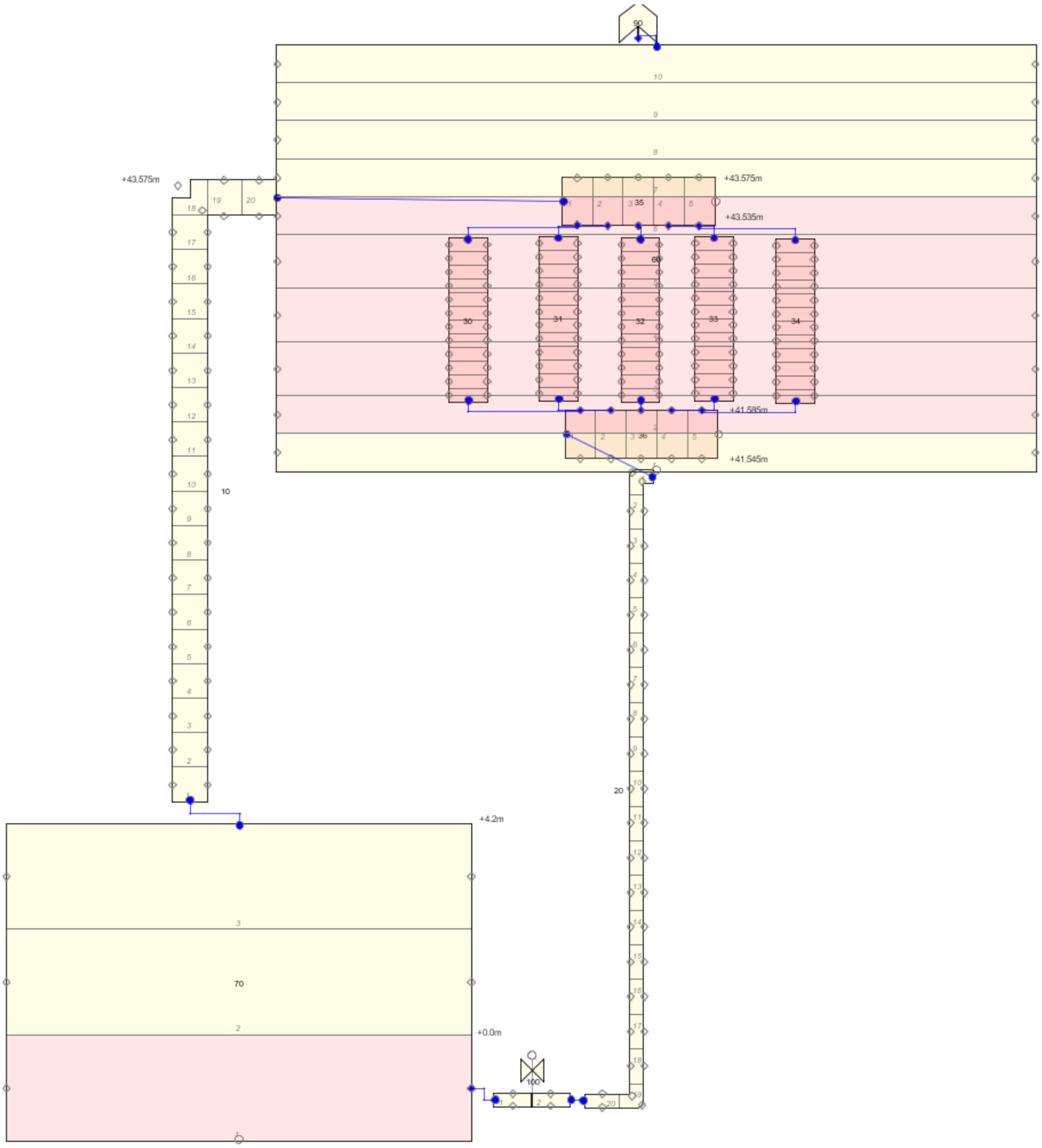


**Figure 25.** TRACE model nodalization of the SPOT PG system.

The AES-2006 has a horizontal type SG with immersed surface for heat exchange. In figure 26 the general view of this SG is presented. In the TRACE model only the secondary side of the SG is modeled. It is modeled with a simple pipe component and the volume of this pipe is set according the real SG of AES-2006. The heat exchanger tubes inside the SG are modeled with simple heat structure elements to bring the correct heat amount to the secondary loop of the SG.

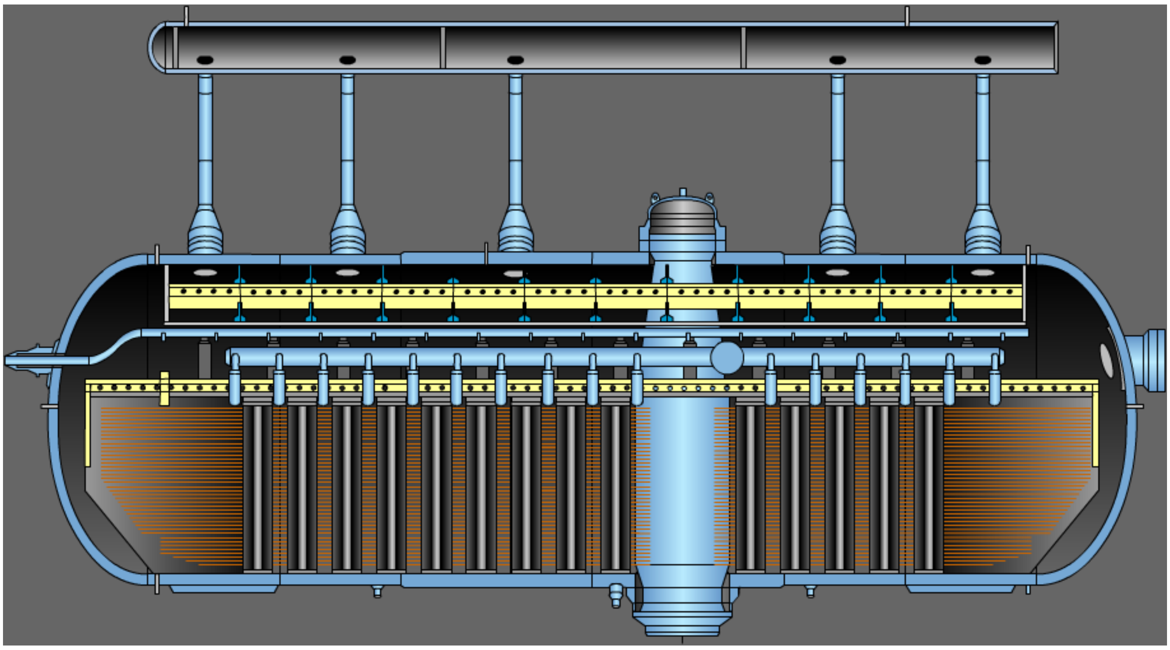

**Figure 26.** General view of AES-2006 steam generator unit [49].

The riser and downcomer sections of the model are modeled with PIPE components. The diameter and the height of these components are set according to the available data of the SPOT PG system (from table 2).

The heat sink water tank in this model is assumed to be only a boundary condition for the heat exchanger-condenser. The tank is modeled with a simple pipe component and the volume of the tank is set so large that the temperature of the water inside the tank is almost constant during the calculation. The tubes and the headers of the heat exchange-condenser are connected to the tank by heat structure components to depict the heat transfer between the water tank and the steam inside the tubes. The break component is used to keep the pressure of the tank constant.

The heat exchanger-condenser model has simplified piping in comparison to the real SPOT PG system. The whole tube bundle of the SPOT PG system (140*16 tubes) is lumped to five pipe components in the model. The length of the tubes is calculated according to the total heat transfer surface of the tubes (the length of the tubes is not same as the distance of the headers (see figure 19). The diameter and the length of the headers are estimated according to the figure 19 (the diameter of the headers is approximately 6.5 times bigger than the diameter of the tubes).

The irreversible pressure losses for the flow area changes of the loop were internally estimated by TRACE code (e.g. connections between the SG secondary side volume and the riser/downcomer piping and the connections of the heat exchanger-condenser and the riser/downcomer piping). The form loss coefficient for the 90° pipe bends is set 0.6 (k-factor) [50]. .

### 6.3 Initial conditions for calculations

The initial pressure of the SPOT PG system calculations is set to 70 bars and the water inside the SG is in the saturation temperature of 559 K. The water capacity in the SG is set according the real value of the AES-2006 SG [51]. The upper part of the SG, the riser, the heat exchange-condenser, and the downcomer was full of saturated steam.

The heat sink water tank was almost full of water at the temperature of 293 K. During the calculations this temperature is almost constant because the amount of water in the tank is very huge (the heat capacity of the water tank is considerably higher than the heat removed from the SPOT PG system). The pressure of the tank was constant 1 bar during the calculations.

The power passed from the primary to the secondary side was changing boundary condition in the calculation matrix. However, the selected power level is kept constant during each calculation. The decay heat power amount of the nominal level is about 6.5% at the moment of reactor shutdown, 1.5% after an hour, 0.4% after a day and 0.2% after a week [52]. In these calculations, it was decided to run five calculations with the decay heat power levels of 6.5%, 2.5%, 1.5%, 0.4% and 0.2% of the nominal level. The nominal thermal power of the AES-2006 NPP is 3200 MW. In addition, in these calculations the SPOT PG system performance is tested in cases of failure of one, two or three loops out of four. With these assumptions the following calculation matrix for the power boundary condition is developed (see table 7).

Table 7. Calculation matrix with different power boundary conditions.

| | | Number of SPOT PG systems | | | |
|---|---|---|---|---|---|
| | | 1 | 2 | 3 | 4 |
| Percentage of the nominal decay heat power | 6,5 % | 208 MW | 104 MW | 69,3 MW | 52 MW |
| | 2,5 % | 80 MW | 40 MW | 26,65 MW | 20 MW |
| | 1.5 % | 48 MW | 24 MW | 16 MW | 12 MW |
| | 0.4 % | 12,8 MW | 6,4 MW | 4,267 MW | 3,2 MW |
| | 0.2 % | 6,4 MW | 3,2 MW | 2,133 MW | 1,6 MW |

### 6.4 Calculation results

The natural circulation system is difficult to calculate accurately, because there is a vast number of factors, which can influence on the phenomenon and has to be taken into account. Nevertheless, even a simplified model, which was created with the TRACE code, has significant predictive capability which can be used to obtain indication of system performance. It seems that the carried calculations of the natural circulation of the SPOT PG system shows satisfactory results for decay heat removal from the AES-2006 reactor. However, the experimental results of the system should be needed to verify and validate the calculation capability of the model.

Each calculation was carried out until the stable natural circulation conditions were reached. Table 8 contains the final values of the calculations. In figures 32-34 the pressure of the loop, the natural circulation of the loop, and the water level of the downcomer as a function of SG power is presented.

**Table 8.** Results of SPOT PG calculations with varied amount of loops.

| 1 loop calculation | | | | |
|---|---|---|---|---|
| Thermal power for every loop (MW) | Percent of decay heat | Pressure in the loop (MPa) | Mass flow rate (kg/s) | Water level (m) |
| 6,4 | 0,2 | 0,22 | 3,0 | 6,0 |
| 12,8 | 0,4 | 0,32 | 6,0 | 9,9 |
| 48,0 | 1,5 | 0,89 | 27,8 | 27,8 |
| 80,0 | 2,5 | 2,00 | 50,1 | 41,2 |
| 208,0 | 6,5 | 22,50 | 38,8 | 42,8 |
| 2 loops calculation | | | | |
| Thermal power for every loop (MW) | Percent of decay heat | Pressure in the loop (MPa) | Mass flow rate (kg/s) | Water level (m) |
| 3,2 | 0,2 | 0,17 | 1,7 | 4,5 |
| 6,4 | 0,4 | 0,22 | 3,1 | 6,0 |
| 24,0 | 1,5 | 0,51 | 11,3 | 13,7 |
| 40,0 | 2,5 | 0,77 | 19,3 | 23,0 |
| 104,0 | 6,5 | 21,80 | 46,0 | 42,8 |
| 3 loops calculation | | | | |
| Thermal power for every loop (MW) | Percent of decay heat | Pressure in the loop (MPa) | Mass flow rate (kg/s) | Water level (m) |
| 2,1 | 0,2 | 0,14 | 1,5 | 4,2 |
| 4,3 | 0,4 | 0,19 | 2,1 | 4,8 |
| 16,0 | 1,5 | 0,51 | 7,4 | 8,8 |
| 26,7 | 2,5 | 0,55 | 12,2 | 11,5 |
| 4 loops calculation | | | | |
| Thermal power for every loop (MW) | Percent of decay heat | Pressure in the loop (MPa) | Mass flow rate (kg/s) | Water level (m) |
| 1,6 | 0,2 | 0,13 | 1,5 | 4,2 |
| 3,2 | 0,4 | 0,17 | 1,7 | 4,5 |
| 12,0 | 1,5 | 0,31 | 5,5 | 8,0 |
| 20,0 | 2,5 | 0,44 | 9,2 | 9,3 |
| 52,0 | 6,5 | 0,97 | 32,7 | 32,4 |

As can be seen from the figure 27 the pressure of the SPOT PG system increased when the decay heat power increased, as expected. Also, the less number of systems in operation, the higher was the pressure of the operating systems. In the cases where 3 and 4 SPOT PG systems were in operation the decay heat power was successfully removed from the primary side to the heat sink tank with all decay heat power levels (from 0,2 to 6,5 %). In the cases where 1 and 2 SPOT PG systems were in operation the pressure of the system increased too high at the decay heat power level of 6,5 % because the heat transfer capacity of the heat exchanger-condenser was not sufficient to take away the sufficient amount of heat from the primary side to the heat sink. In TRACE model the relief valve of the SG was not modeled so the pressure increased until the fatal error of the code cut off the calculation. In real NPP the pressure relief valve opens at some pressure level and depressurizes the system. However, the SPOT SG system should be dimensioned such that it can remove enough heat – e.g. up to 2.5 % decay heat – without overpressurising the SG.

The reason for the pressure increase in the system was insufficient condensation rate in the heat exchanger-condenser. The increased pressure should also improve the condensation of the steam but it seems that this only decelerates the pressure increase. The calculation cases where the pressure increased too high and the calculation did not reach a stable condition before error of the code are removed from figures 27-29.

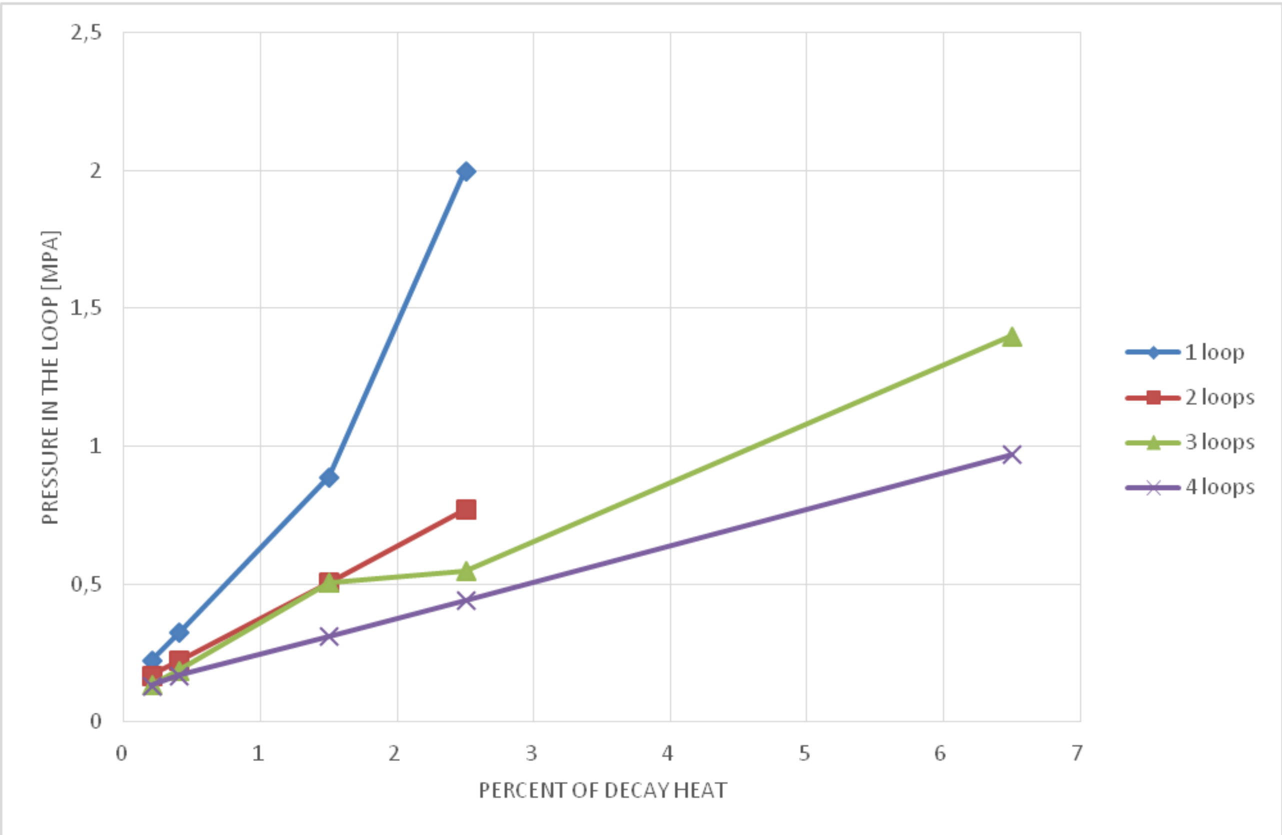


**Figure 27.** Relation between the decay heat power and the pressure in the loop.

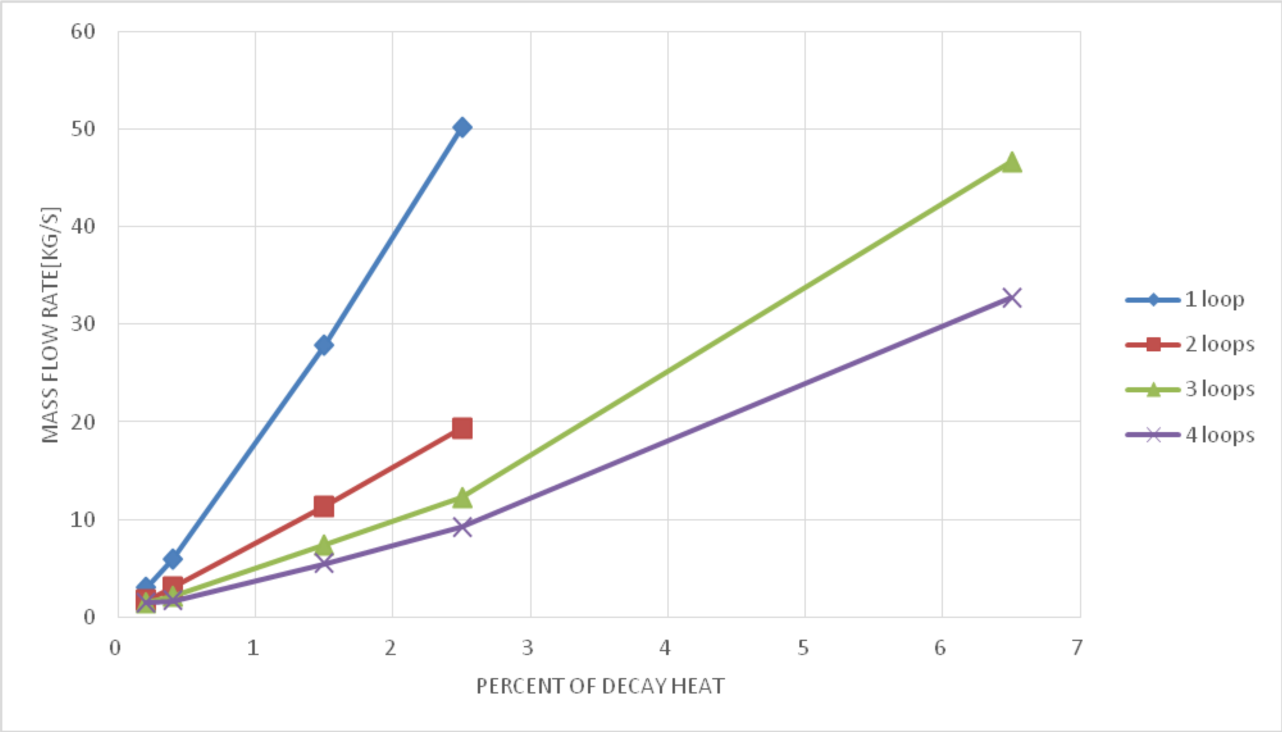


**Figure 28.** Relation between the decay heat power and the mass flow rate of natural circulation in the loop.

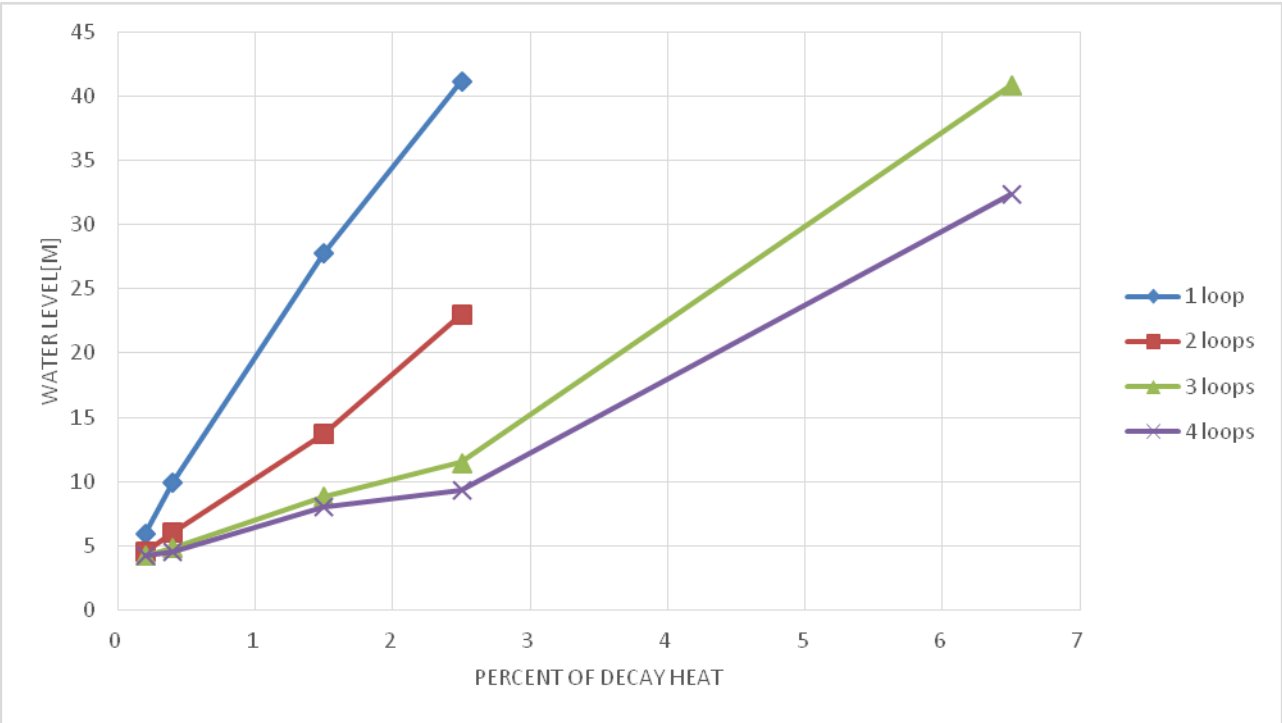


**Figure 29**. Relation between the decay heat power and the water level in the downcomer

The higher decay heat power leads to the greater mass flow rate of the steam in the riser (figure 28). Reason for this is the higher boiling rate in the SG because of the higher power. The higher decay heat power was, the higher water level in the downcomer was. In most cases the water level was not so clear because the downcomer was partly filled with steam.

According to these calculations, it seems that at least three SPOT PG systems are needed to take away core decay heat effectively from the primary side to the heat sink. This is consistent with the information from the official report that each of the four SPOT PG systems has capacity of 33.3% [22].

# CONCLUSIONS

The study was set out to explore a role of passive safety systems in modern and existing NPP concepts and reveal potential problems of such systems.

Passive safety systems are included in modern reactor concepts and there are many reasons for this. One of the most important ones is that passive system helps to decrease the capital costs of the NPP because it does not require energy sources and does not contain moving parts. Furthermore, they do not require control signals, nor actions of operating personnel. The NPP concepts of new generation with an extensive use of passive safety systems, which do not need electricity to function, would have prevented the accident on the Fukushima NPP, where loss of electric supply caused a loss of cooling function. Also the passive safety systems are used due to the fact that they can help to reduce the need for action from the people operating the plants.

The passive safety systems of advanced reactors like AES-2006 have ability to work autonomously for periods up to 24 hours and much longer. It allows operators to do more actions during the accident to restore the plant to safe state.

Passive safety systems have some disadvantages, which are connected with various failure modes leading to failure of natural circulation phenomenon. Natural circulation has lower driving forces than active safety systems. Therefore, the most of the possible failure modes are connected with a low driving force of natural circulation and possible termination of such phenomenon. In addition, a thermal stratification is another issue that requires special attention.

The accurate ways to test the performance of the safety systems are to carry out experiments on research facilities and to make a computational research of the system in question. Such approaches allow exploring the behavior of the system in various regimes, to simulate design basis accidents, to identify failure modes, and evaluate the probability of the failure. Experiments on facilities may cost a lot, but on the other hand they can represent system behavior more realistically than calculations. Such facilities, like large-scale GE2M-PG and KMS for AES-2006, allow to simulate the whole process of heat removal (outside the containment and outside the core) and take into account factors which are hard to model during computer calculations.

Nevertheless, with computer calculations it is also possible to receive important results and check the performance of the passive safety system. In chapter 5 the work of the SPOT PG system was tested and the data obtained appeared to be similar to the information from the reports. It shows that three loops (out of four) of SPOT PG system have enough capacity to provide the necessary level of safety (removing decay heat of 6.5 % at system pressure well below the SG normal operating pressure).

It is worth saying that there are literally no prerequisites to abandon application of passive systems, but prevention of numerous failure modes is one of the complicated challenges for nuclear community. The whole nuclear community aims to increase the safety of new nuclear technologies today. The accurate ways to test new features are to carry out experiments on large-scale facilities and to make a computational research of the system in question. Such approaches allow exploring the behavior of the system in various regimes, to simulate design basis accidents, to identify failure modes, and evaluate the probability of the failure.

Thus, the modern reactor concepts instill confidence that in the future passive safety systems will evolve toward replacement of active systems and greater autonomy.

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