

ИССЛЕДОВАНИЕ АВАРИЙНЫХ СИТУАЦИЙ С ПОТЕРЕЙ ТЕПЛОНОСИТЕЛЯ НА АЭС: ВОПРОСЫ БЕЗОПАСНОСТИ ВВЭР-1000

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Принятые процедуры и нормы не являются полностью адекватными, если выполняются просто механически. Уроки трех крупных ядерных аварий, в том числе, на Три-Майл-Айленд, в Чернобыле и на Фукусима Дайичи, продемонстрировали недостаточность существующих методов, стандартов и правил для предотвращения аварий. Поэтому постоянно ведется работа по предотвращению повторения таких бедствий или минимизации их последствий. Многие критерии и проекты были предложены и использованы в новых поколениях АЭС. В настоящей работе основное внимание уделяется изучению аварий с потерей теплоносителя и различным способам работы ученых и инженеров по её предотвращению и безопасному удержанию теплоносителя.

Ключевые слова: АЭС, ВВЭР-1000, ядерная безопасность, авария с потерей теплоносителя.

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INTRODUCTION

In water cooled reactors there are two types of potential accidents in terms of safety analysis [1]: loss of coolant accidents (LOCA) and transients. LOCA refer to “those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system” [3] while transients are events that causes reactor trip without loss of coolant as a result of human error or equipment failure including main circulation pump. These accidents have radiological consequences to the public if not contained and controlled appropriately. VVER-1000 is equipped with emergency core cooling system (ECCS) and containment spray systems to solve any event that will cause pressure drop beyond design including LOCA and transient. These systems have 3x100% redundancy with the exception of ECCS water storage tank [2] located under the containment.

The lessons learnt from major nuclear accidents including “Three Mile Island” in USA, the Chernobyl NPP in USSR and “Fukushima” NPP in Japan, are now considered in the safety design of NPPs. The Three Mile Island on March 28, 1979 involves Pressurized Water Reactor (PWR) of western type, the Chernobyl disaster on April 26, 1986 involves RBMK (Реактор Большой Мощности Канальный) and Fukushima Daiichi accident on March 11, 2011 involves a Boiling Water Reactor (BWR). 67 reactors have been constructed in different parts of the world since 1960 [3] and VVER has no record of any major accident.

TYPES OF LOCA

There are two types of LOCA: large break LOCA and small break LOCA [2]. For VVER-1000, the initiating event for a large LOCA is a guillotine break of the reactor main circulation pipeline with a diameter of 850 mm at the reactor inlet while the initiating event for small break

LOCA is a small leak in the primary pipeline or rupture of the main circulation pipeline with equivalent diameter of less than 100 mm.

Reactor will go subcritical immediately in case of large break LOCA due to blow-out of large quantities of coolant leading to a pressure drop in the primary circuit and reduction in neutron moderation. Loss of unit power, tripping of the main circulation pump and loss of steam removal to turbine will occur with dire consequences on the cladding material (zirconium alloy) with design maximum temperature of 355°C. To avoid fuel meltdown, a reactor protection system (RPS) is actuated leading to flooding of the reactor core by ECCS and containment spray system. Considerable thermal energy will still be generated in the fuel from decay of radioactive fission products after shutdown with generation rate of 7% of the thermal power during normal operation [1]. The failure of the RPS will result in energetic chemical reactions between cladding material and residual water-steam in the reactor pressure vessel as a result of high temperature. This reaction leads to generation large quantities of hydrogen in the reactor core, melting of the core and release of the fission products to the containment and possibly to the environment.

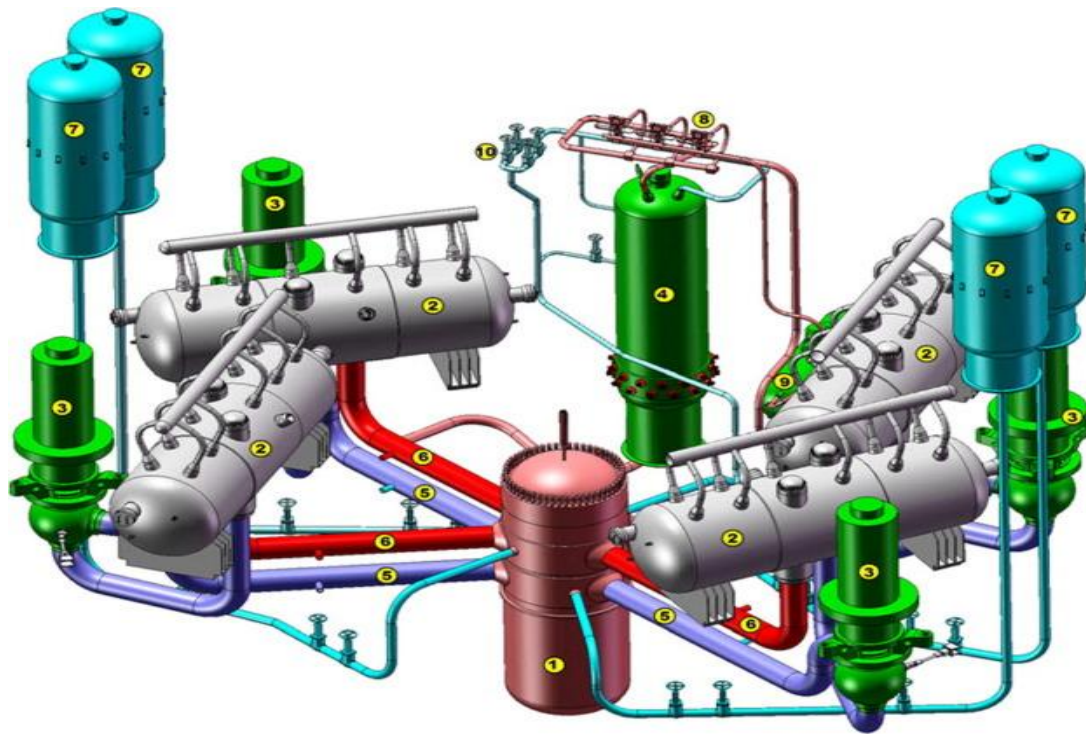


Fig. 1. Primary circuit of VVER-1000 plant; (1) reactor, (2) SG, (3) main coolant pump, (4) pressurizer, (5) cold leg, (6) hot leg, (7) accumulator, (8) PRZ pulse safety device valve, (9) relief tank, (10) injection system [4].

Safety systems should be designed to withstand the adverse effect of the environment during accidents and should follow the criterion stated in [1] that “structures, systems, and components important to safety must be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss of coolant accidents.”

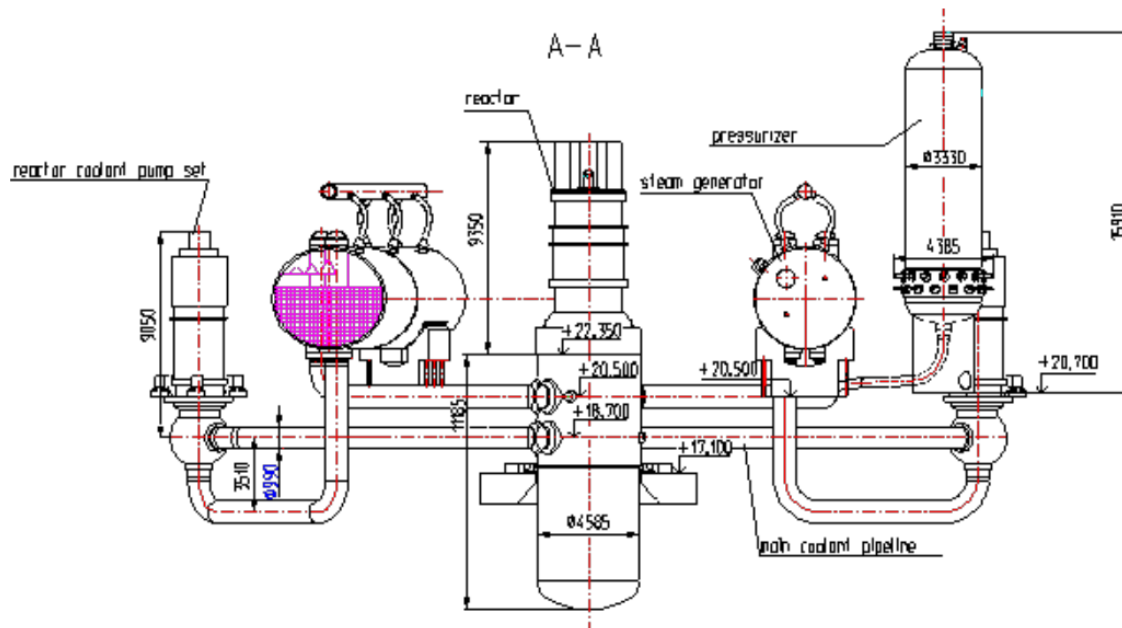


Fig. 2. Layout of the reactor coolant system

The function of the containment spray system reduces containment pressure and temperature under LOCA. It should also remove radioactive fission products from the containment atmosphere in order to prevent the accumulation of combustible gases including hydrogen. The system of hydrogen removal is to prevent uncontrollable hydrogen ignition thereby adding to the problem of LOCA.

The function of the ECCS is to inject water into the core (via spray and/or flooding systems) to remove heat from the core in the event of design basis accidents. It includes four (4) channels with each having two emergency makeup pumps and a delivery pipeline (nominal diameter of 100mm) supplying borated water into the non-disconnected parts. Its consideration include isolation from RCS, redundancy to meet reliability targets and safety function within the prescribed limits for accident conditions. ECCS consists of the following:

- a. High pressure emergency cooling system
- b. Passive emergency core cooling system
- c. Low pressure emergency cooling system

The high pressure emergency cooling system is responsible for supplying borated water to the primary circuit under LOCA conditions and leaks from the secondary circuit including the steam generator. Passive ECCS is responsible for quick supply of boric acid to the reactor under LOCA when pressure drops to below 6MPa. If the pressure drops to 2.5MPa, the low-pressure emergency cooling system will supply borated water to the primary circuit for reactor core cooling.

SEQUENCE OF EVENTS DURING A LOCA

A typical LOCA sequence in a light water reactor including VVER-1000 is well documented in [5]. It has the following phases:

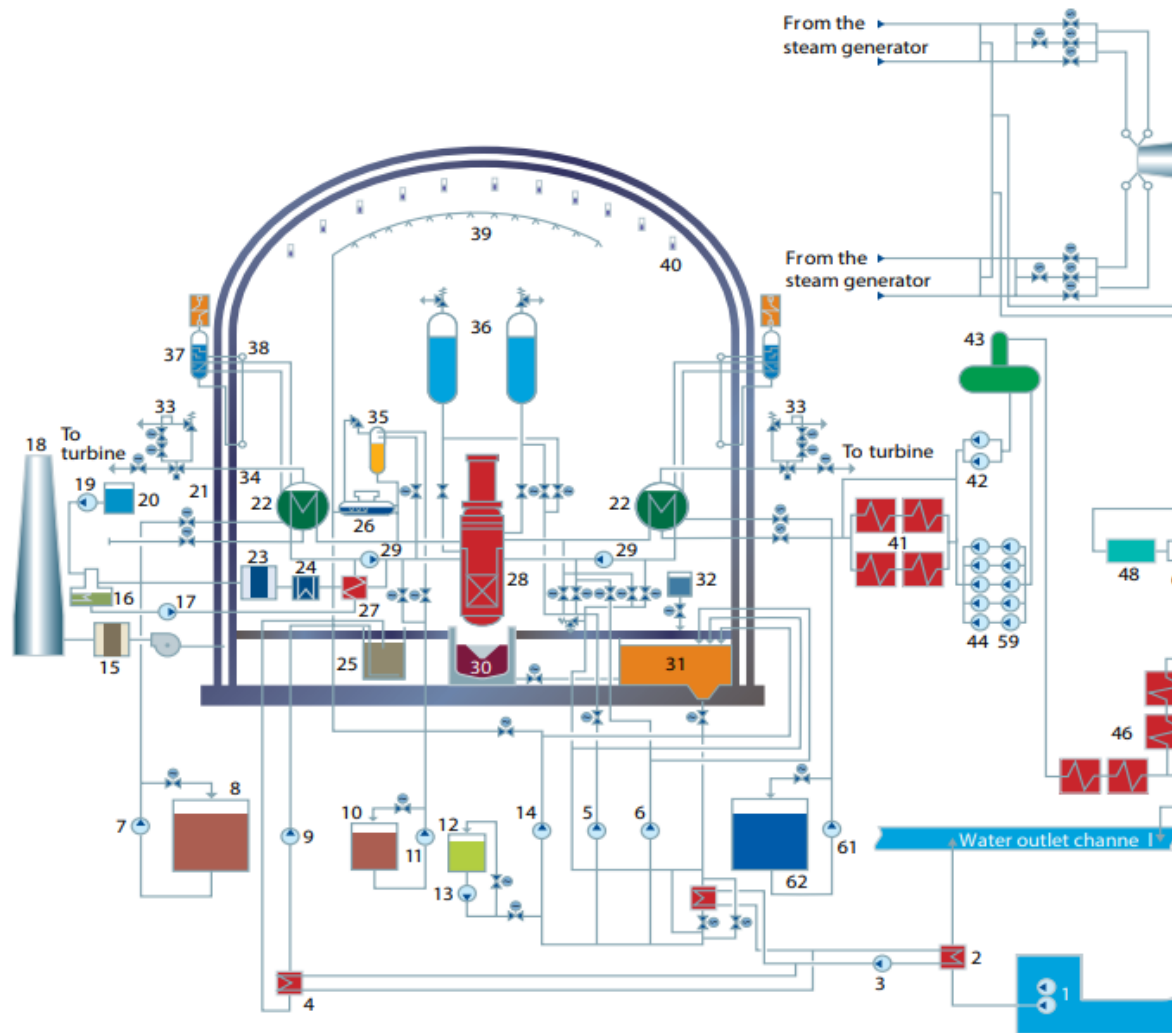


Fig. 3. Layout of VVER-1000 reactor building. 1-service water pump, 2-intermediate cooling circuit heat exchangers, 3-intermediate circuit pump, 4-spent fuel pool heat exchanger, 5-emergency injection system, low pressure pump, 6-emergency injection system, high pressure pump, 7-emergency feed water pump, 8-storage tank for high concentration boric acid, 9-spent fuel cooling pump, 10-storage tanks for boric solution, 11-emergency boration system pump, 12-storage tank for chemical reagent, 13- supply pump for chemical reagents, 14-containment spray system pump, 15-filter, 16-volume and chemical control system deaerator, 17-volume and chemical control system pump, 18-ventilation stack, 19-controlled-leak pump, 20-controlled-leak tank, 21-external containment, 22-steam generator, 23-water-treatment plant, 24-after-cooler, 25-spent fuel pool, 26- bubble tank, 27-regenerative heat exchanger for volume and chemical control, 28-reactor, 29-reactor coolant pump, 30-core catcher, 31-emergency core cooling system sump and refueling water storage, 32-alkali (NaOH) emergency reserve tank, 33-MSIV, safety and relief valve assembly, 34-containment, 35-pressurizer, 36-ECCS hydroaccumulators, 37-passive heat removal system tank, 38-condenser for the containment passive heat removal system, 39-spray system, 40-passive hydrogen recombiner, 41-high pressure heaters, 42-electric-powered auxiliary feed water pump, 43-deaerator, 44-electric-powered feed water pump, 59-boost pump, 60-condensate pumps for unit demineralization plant, 61-emergency feed water pump, 61-emergency feed water pump

1. Depressurization of the primary system followed by an automatic scram. This will trigger the oscillation of the fuel temperature depending on the position, power reduction time and depressurization time. Due to pressure decrease of water the temperature of the water and that of the fuel cladding will also decrease because of the cooling capacity of water.

2. Heat-up of the core caused by heat generation (decay heat) in the fuel and loss of water inventory. The rate of heat-up varies with fuel design, operating power at the time of the LOCA and plant design. When the temperature exceeds 600°C, the cladding material will be prone to plastic deformation and may balloon under the effect of the inner pressure of the rod. The fuel rod will burst open, breaking the first barrier to radiation containment, if the ballooning exceeds a certain size. The zircaloy cladding can undergo phase transition from α phase to β phase at temperature exceeding 900°C.

3. The cladding temperature will continue to increase gradually due to increase in radiative and convective heat transfer. This is reversed under the ECCS effect and a period of gradual temperature decrease, the cladding will cool down to comfortable level due to rise of ECCS water in the core.

ACCEPTANCE CRITERIA FOR LOCA

The acceptance criteria of a LOCA for PWRs [6] include:

1. The fuel rod cladding temperature should not exceed the prescribed value (typically 1200°C).
2. The maximum local cladding oxidation should not exceed a prescribed value (typically 17-18% of the initial cladding thickness before oxidation).
3. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed a prescribed value (typically 1% of the hypothetical amount that would be generated if all the cladding in the core were to react).

The purpose of the first two criteria is to ensure that the cladding will remain sufficiently ductile so that it does not shatter into pieces during and after the quench phase of the LOCA transients or the fuel cladding entering into a regime of runaway oxidation and uncontrollable core heatup [2]. These criteria were violated in the accident of Three Mile Island in 1979.

The acceptance criteria for all accidents leading to containment depressurization in addition to that of LOCA include:

1. The calculated peak containment pressure needs to be lower than the containment design pressure and the containment pressure needs to be higher than the corresponding acceptable value.
2. Differential pressures acting on containment internal structures have to be maintained at acceptable values.

LONG TERM PRESSURE-TEMPERATURE TRANSIENTS

The pressure-temperature behaviour of the post-accident containment atmosphere analyzed for a time period of several hours results from an imbalance between heat sources and heat sinks [5 loca iaea]. The heat sources include reactor residual heat released, the accumulated heat in structures of both primary and secondary circuits and heat released by potential hydrogen burning or explosions. The heat sinks include heat absorption in the containment walls and other heat absorbing structures, spray system operation, fan and vent cooler operation, pressure suppression heat sinks, and ECCS operation.

The acceptance criteria include:

1. the excess pressure in the containment can lead to radioactivity releases through containment leakages therefore, the calculated doses should be below the limits for design basis accidents (DBA).
2. The acceptance criteria for LOCA stated above.
3. There should be no failure of the containment.

4. There should be no immediate health hazards on the population
5. The effects of ^{137}C release limit should not exceed the prescribed value (100TBq) and the other nuclides should not cause any danger after the specified period.

CALCULATION OF HYDROGEN GENERATION AND PEAK CLAD TEMPERATURE

As stated in the safety criteria for LOCA, the calculated total amount of hydrogen generated from chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that will be generated if all the metal in the cladding surrounding the fuel were to react. The hydrogen released when zircaloy is oxidized in steam is directly proportional to the oxygen consumed or zircaloy reacted. The amount of hydrogen produced at any time is expressed as

$$W_H = 2 * \frac{M_H}{M_{Zr}} Kpt^{0.5}$$

Where W_H is the mass of hydrogen produced per unit (cm^2) of zircaloy surface, M_H is molar mass of hydrogen gas (2.016) mg/mg mol, M_{Zr} is molar mass of zircaloy (91.22) mg/mg mol, t is the time in seconds, and Kp is the parabolic rate constant (mg Zr/cm^2) $^2\text{s}^{-1}$. If the oxidation does not deviate significantly from a parabolic rate, $Kp = 3.33 \cdot 10^5 \cdot \exp(-140600/RT)$.

The LOCA peak cladding temperature limit limits the maximum pellet linear power density [1267]. The LOCA linear power density limit on the pellet for PWRs is expressed in terms of the total peaking factor defined as

$$F_Q = \frac{\text{Power}_{\text{peak}}^{\text{pellet}}}{\text{Power}^{\text{pellet}}} = \frac{\frac{\text{Max}_{x,y,z} P(x,y,z)}{V_{\text{core}}}}{\frac{1}{V_{\text{core}}}} = \frac{\text{Max}_{x,y,z} P(x,y,z)}{\int_{V_{\text{core}}} P(x,y,z) dx dy dz}$$

Which in practice is likely to have an axial dependence due to LOCA blowdown and reflooding characteristics obtained by completing the numerator's maximization over x, y .

SAFETY SYSTEM AND DISTINCTIVE FEATURES OF VVER

The safety systems of VVER include the protective, localizing, supportive and control safety systems which are provided for preventive or to limit reactor damage and localization of radioactive substance within the NPP [2313]. The technical data for “Tianwan” NPP design with V-428RP are given below in tables (1) and (2).

Tab. 1. Characteristics of high-pressure emergency injection pump

Parameter	Value
Nominal capacity, t/h	150
Pressure head at nominal supply, MPa	6.5
Maximum capacity, t/h	260
Pressure at maximum capacity, MPa	3.5
Boric acid concentration, g/kg	16
Operating temperature of medium, °C	70-95

Tab. 2. Characteristics of low-pressure emergency boron injection

Parameter	Value
Nominal capacity, t/h	800

Pressure head at nominal supply, MPa	1.5
Maximum capacity, t/h	900
Pressure at maximum capacity, MPa	1.2
Operating temperature of medium, °C	70-95

The passive part of ECCS consists of eight (8) hydroaccumulators (HA-2), combined in four (4) groups. The structure of each group involves two 2nd stage hydroaccumulators (volume 120m³ each), pipelines with valves [iris]. The 2nd stage hydroaccumulators are connected via discharge line to the pipings of the 1st stage, boron solution is supplied from hydroaccumulators into pressure and collection chambers of the reactor. Two of them are connected to the reactor vessel to supply water for flooding the bottom of the core while the remaining two are used to flood the top of the core. When the pressure in the reactor decreases below that in the hydroaccumulator, the check valves open and a working medium is pressed out into the reactor due to pressure of nitrogen blanket. Nitrogen into the reactor is prevented by the isolation of hydroaccumulator with quick-acting valves when the level drops below the permissible value.

The emergency gas removal system serves to remove steam-gas mixture from the primary circuit and decrease the pressure below DBA. It can be used by the operator under normal operating conditions and DBA. The system consists of a set valves, pipelines connected to the primary circuit for removal of steam-gas mixtures into the relief tank from reactor top head, from primary collectors of steam generators, and from pressurizer.

The over-protection system is installed on the primary and secondary circuit. It consists of three pilot-operated relief valves independent of each other and connected to the pressurizer. The pressurizer provides implementation of “feed and bleed” procedure to decrease the primary pressure to 1MPa under DBA caused by melt-through of the reactor.

Localizing systems serve to prevent or limit the spread of radioactive substances released during accidents inside the NPP and into the environment. The design value of a leak into space between the confinement wall and the external envelop after a hypothetical accident is 0.2% of the air mass in the building for 24 hours. The design pressure and temperature of the confinement are 0.5MPa and 150°C respectively. The corium retention and cooling system serves to retain liquid and solid fragments of the damaged core, reactor vessel parts, internals under severe accident involving reactor vessel melt-through and core melting. There is the sprinkler system within the localizing system whose function includes:

1. Pressure decrease after the design basis accidents
2. Post-accident decay heat removal from containment
3. Control of water temperature in the containment sump
4. Removal of radioactive aerosols and iodine from containment

The supporting systems serve to provide safety systems with power, working medium and conditions for their operation. The supporting systems include service water systems for essential consumers and emergency power supply system. An intermediate cooling circuit system of essential consumers serves as a barrier between the auxiliary systems of the reactor plant containing radioactive media and service water system.

CONCLUSION

The study of LOCA as a panacea for accidents in nuclear power plants has been done. There are no documented cases of LOCA accident involving pressurized water reactor (PWR) of

the Russian type (VVER). Experience has shown that safety is not one directional (machineries alone) but include the personnel working at the power plants. The three major nuclear accidents were partly due to human misjudgment of the situation and LOCA. Therefore, the design of new generation power plant understudy the lessons learnt from these experiences, the researches undertaken by scientists and engineers, the minimum safety criteria to be utilized by international (IAEA) and national regulators, operators and consumers to design now more robust and safety-oriented NPPs.

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A Study of Loss of Coolant Accident (LOCA) in NPPs: a Case of Safety in VVER-1000

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Abstract – Sound procedures and good practices are not fully adequate if merely practiced mechanically. The lessons of the three major nuclear accidents including Three Mile Island, Chernobyl, and Fukushima Daiichi have displayed insufficiency of existing techniques, standards and rules for their forestalling. Therefore, studies are constantly undertaken to militate against a repeat of such disaster or to limit it. Many criteria and designs have been proposed and utilized in new generations of NPP. This paper focuses on the study of loss of coolant accident (LOCA) and the various ways scientists and engineers are working to avoid and safely contain it.

Keywords: NPP, VVER-1000, nuclear safety, coolant accident.