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# Comparative Transient Analysis of Large Break Loss of Coolant Accident (LBLOCA) with Subsequent Loss of Offsite Power and all AC power in VVER-1200 Reactor using PC Based VVER-1200 NPP Simulator

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## Abstracts

The generation III+ VVER-1200 (AES-2006) reactor design is equipped with improved safety features with component redundancy compared to the previous models. This paper is concerned with the performance of the active and passive safety features during the transient of major Beyond Design Basis Accident (BDBA) scenario. The study is done considering Large Break Loss of Coolant Accident (LB LOCA) with Loss of Offsite Power (LOOP) and LB LOCA with loss of all AC power (LOACP) sources namely station blackout using PC based VVER-1200 NPP simulator. By the effectiveness of the active and passive cooling components, the temperature of the hot leg and cold leg is observed to be 104 °C and 69 °C for LB LOCA with LOOP and 135 °C and 125 °C for LB LOCA with LOACP. Comparative analyses are compatible with IAEA safety guideline.

**Keywords:** VVER; LB LOCA; LOOP; LOACP; BDBA

## Introduction

It has been a quest for mankind to design a safe, reliable, and efficient nuclear reactor. As a result, different kinds of reactors are designed based on different requirements. A nuclear reactor can be defined as a system where a fission chain reaction is initiated, controlled, and sustained. Thus, nuclear reactors can be classified based on different reference parameters. The most commonly known reactors are Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR). There are also Graphite Cooled Reactors (GCR) where graphite is used as moderator, Pressurized Heavy Water Reactor (PHWR), Fast Breeder Reactor (FBR) where fast neutrons initiate fission [1,2]. According to the International Atomic Energy Agency (IAEA), there are 298 PWR reactors, 73 BWR reactors, and a total of 451 nuclear reactors are operating throughout the globe [3]. The design of any nuclear reactor is equipped with a high level of safety prescribed by IAEA safety guideline [4, 5].

Despite the presence of safety arrangements, the world has experienced some of the biggest nuclear disasters. Of them, the most noteworthy are the Three Mile Island Accident (1979), Chernobyl Disaster (1986), and The Fukushima Daiichi Nuclear Disaster (2011). There are many reasons for these accidents like human error, design flaws, natural disasters etc. But the consequences were dangerous, nevertheless. This led to the death of many people living near the site by getting the radiation dose [6,7].

It has been seen that most of the major accidents were caused by Loss of Coolant Accident (LOCA), Loss of Offsite Power (LOOP), and Loss of AC Power (LOACP). Among them, LOCA is studied extensively[8-10]. LOCA is the loss of coolant from the reactor due to rupture in the main coolant pipeline. There are two kinds of LOCA, depending on the size of the leak or rupture. A Large Break LOCA (LB LOCA) is a double-ended rupture in the primary coolant pipeline with a leak greater than 0.1 m2 [11-15]. The scenario of loss of offsite power represents the failure of active part components which need electricity from the grid and failure of the active part as well as backup diesel generators in the case of loss of AC power.

Most of the severe accidents occur due to cooling system failure. It is very difficult to predict a severe accident scenario. Before an accident, there are some initiating events like loss of coolant accident (LOCA), Total Loss of Feed Water (TLOFW), Station Black Out (SBO), and Steam Generator Tube Rupture (SGTR). The comparative analysis of LBLOCA with LOOP and LBLOCA with LOACP using VVER-1200 NPP simulator is essential task as these events may arise in any moments. Proper knowledge and management may help to overcome such kinds of disastrous situation. Present work represents the investigation of safety systems failure of VVER- 1200 reactor in worst accident case and its consequence. This simulation research will be helpful to understand the new coming VVER-1200 Power reactor operating manner and also help to develop skilled manpower.

## Technical Specification of the Simulator

The simulator is supplied by Western Service Co. (WSC), US with 3KEYMASTER™ modeling tools which include 3KEYMASTER™ instructor Station. The simulator covers the full range of plant operations from plant cold shutdown to hot standby, hot zero power, and to full range of power maneuvers as well as all possible transients. The simulator is intended to simulate the VVER-1200/V392M technology. It works as real-time, full scope, high fidelity simulator. It consists of more than 150 Human Machine Interfaces (HMIs) which represent the technology schemes of NPP. The simulator allows students to perform complete plant startups, shutdowns, load maneuvers, and simulate normal and abnormal plant transients, including malfunction scenarios [16].

The simulator is a Pressurized Light Water Reactor with four circulation loops, four steam generators and four reactor coolant

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transfer pumps. The Reactor total thermal power is approximately 3200 MW and the turbine electric power is approximately 1200 MW. The steam generators supply steam to the turbine which consists of high-pressure turbine and four low pressure turbine cylinders. Condensate is pumped from three condensers by the condensate pumps through low pressure heaters to deaerator and to the suction of five feed water electric pumps. The Feed water pumps provide necessary feed flow to the steam generators through high pressure heaters. Four moister separators re heaters separate moisture from high pressure turbine exhaust steam, reheat it in two stages re heaters and direct steam to the low pressure turbines. Table 1 contains the primary equipment and specifications that are the basis for the simulator development. [16]

| **Plant Item Reactor** | **Plant Description** |
|---|---|
| Configuration | 4 loops, 4 Reactor Coolant Pumps, 4 Steam Generators |
| Reactor core power (Nominal) | 3200 MW |
| Pressurizer pressure (Nominal) | 161.2 kg/cm2 |
| Hot leg temperature (Nominal) | 328.9°C |
| Coolant inlet temperature (Nominal) | 298.2 °C |
| Average temperature rise in vessel | 30.7°C |
| Coolant flowrate trough reactor vessel | 86000 m3/h |
| Number of control rods clusters | 121 |
| Number of fuel assemblies | 163 |
| Absolute steam pressure at the steam generator | 70 kg/cm2 |
| Steam temperature (Nominal) | 287 °C |
| Containment | |
| Inner cylinder diameter | 50.8 m |
| Height | 64.4 m |
| Maximum design pressure | 5.0 kg/cm2 |
| Maximum design temperature | 150°C |
| Free Volume | 66x103m3 |
| Emergency Core Cooling System (ECCS) | |
| Accumulators number (JNG50-80) | 4 |
| Accumulator nominal pressure (JNG50-80) | 60.0 kg/cm2 |
| Accumulator total volume (JNG50-80) | 240.0 m3 |
| Accumulators number (JNG10-40) | 8 |
| Accumulator nominal pressure (JNG10-40) | 1.0 kg/cm2 |
| Accumulator total volume (JNG10-40) | 960.0 m3 |
| High pressure injection charging pumps number (HP) | 4 |
| HP design flow | 14.5 m3/h |
| Intermediate pressure safety injection pumps number (SI) | 2 |
| SI pump design flow | 230 m3/h |
| SI pump design head | 65 kg/cm2 |
| Residual head removal pumps number (RHR) | 2 |
| RHR pump design flow | 800 m3/h |
| RHR pump design head | 14.7 kg/cm2 |
| Turbine | |
| Configuration | 1 High pressure turbine, 4 Low pressure turbines |
| Generator power (Nominal) | 1200 MW |
| Moister separator reheaters number (MSR) | 4 two stage MSRs |

| | |
|---|---|
| Balance of Plant Systems (BOP) | |
| Condensate pump number | 3 |
| Low pressure heaters (LPH) | 3 LPH#1, 3LPH#2, 3 LPH3, 3LPH#4, |
| Main feed water pumps number | 5 electric pumps |
| High pressure heaters (HPH) | 2 HPH#5, 2 HPH#6, |
| Auxiliary feed pump number | 1 centrifugal pump |

**Table 1:** Primary equipment and specifications.

## Reactor Core Modeling

Following design criteria have been considered for modelling the reactor core [16]:

- The minimum departure from nucleate boiling ratio during normal operation and anticipated operational occurrences is not less than 1.19.
- The maximum fuel centerline temperature evaluated at the design overpower condition is below that value which could lead to centerline fuel melting. The melting point of the UO2 is not reached during normal operation and anticipated operational occurrences.
- Fuel rod clad is designed to maintain cladding integrity throughout fuel life.
- Each reactor system is designed so that any xenon transients will be adequately damped.
- The reactor coolant system is designed and constructed to maintain its integrity throughout the expected plant life.
- Power excursions that could result from any credible reactivity addition incident do not cause damage either by deformation or rupture of the pressure vessel, or impair operation of the engineered safety features.
- The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, reactor power transients remain bounded and damped in response to any expected changes in any operating variable.

Figure 1a and 1b shows the primary and secondary circuit layout of PC Based VVER-1200 NPP simulator [16].

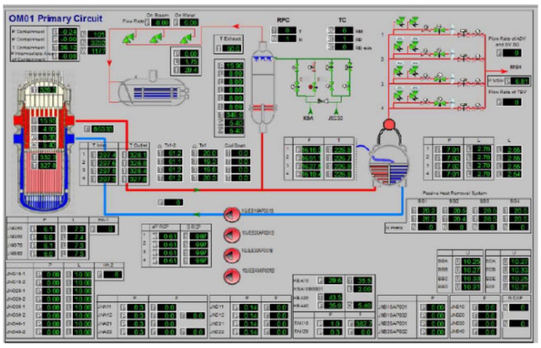


**Figure 1a:** Shows the primary circuit layout.

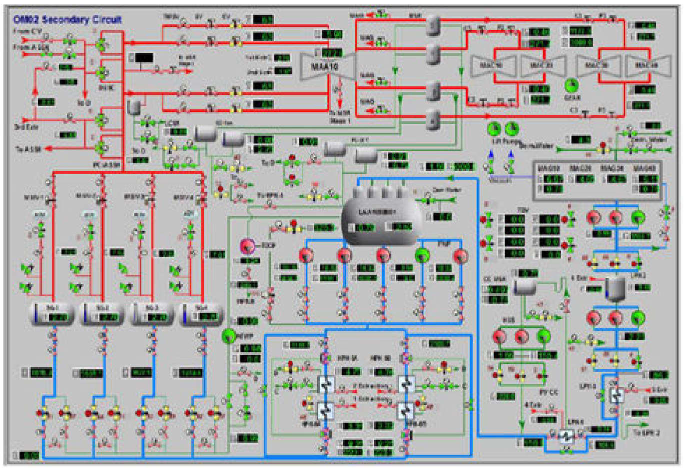


**Figure 1b:** Shows the primary circuit layout.

The simulator offers a simple user-friendly screen layout. The simulator can be operated in real-time or accelerated time mode. The advantage of using a real-time simulator is that user can understand the response of the systems which correctly represents the real system, without delay or limitations as pre-recorded scenarios. The evolution of NPP simulators with real-time is described in [16].

## Process of Simulation

1. To simulate the LB LOCA scenario concurrent with LOOP and LOACP, following initial condition at nominal power level has been considered Table 2.1-2.3.

| Initial condition parameter at nominal power level | Value |
|---|---|
| thermal power of reactor (MW) | 3212 |
| Electrical Power (MW) | 1180 |
| Primary pressure at the core outlet, absolute, MPA | 16.2 |
| Coolant temperature at reactor inlet (°C) | 298.2 |
| Coolant temperature at reactor outlet (°C) | 328.9 |
| Coolant flow rate at reactor inlet, $m^3/h$ | 86000 |
| Pressurizer level, m | 8.1 |
| steam pressure at the steam generator (steam header) outlet, MPA | 7.00 |
| Steam temperature at rated load °C | 287.0 |
| Steam humidity at steam header outlet, %, not exceeding | 0.2 |
| Feed water temperature at rated load °C | 225 |

**Table2.1:** Initial condition parameter at nominal power level.

Following safety systems for reactor emergency core cooling have been incorporated in the simulation:

| Safety systems of VVER1200-power unit | |
|---|---|
| High Pressure active part of Emergency Core Cooling System (ECCS) | Two-channel active system (2 channels of 100% each) |
| Low Pressure active part of Emergency Core Cooling System (ECCS) | Two-channel active system (2 channels of 100% each) |
| ECCS Passive part Hydro-accumulator – 1 (HA-1) | Passive four-channel system<br>(4 channels of 33% each) |
| System of core passive flooding Hydro-accumulator- 2 (HA-2) | Passive four-channel system<br>(4 channels of 33% each) |
| System of emergency boron injection | Two-channel active system (2 channels of 100% each) |
| System of emergency Steam Generator (SG) cooling | Closed two-channel active system<br>channels of 100% each) |

In the simulation, 3 different Initial Condition (IC) have been established based on core fuel lifetime such as Beginning of core life (BOL), Middle of core life (MOL), End of core life (EOL) and core power level:

| IC No. | Core life | Core power (%) | Description |
|---|---|---|---|
| IC-001 | BOL | 100 | Full power IC, core fuel loading is beginning of core life for equilibrium cycle, equilibrium Xe and Sm. |
| IC-002 | MOL | 100 | Full power IC, core fuel loading is end of core life for equilibrium cycle, equilibrium Xe and Sm. |
| IC-003 | EOL | 100 | Full power IC, core fuel loading is end of core life for equilibrium cycle, equilibrium Xe and Sm. |

**Table 2.2:** Initial condition (IC) based on core life and core power level.

Following boundary condition and shutdown setpoints have been implemented for simulating the studied BDBA scenario.

- Short neutron flux period T<10 second
- High neutron flux >107% Nnom
- Low Rx pressure PRX <14.7 MPA and power >75%
- High T-hot Thot>330°C and P<137.2 bar
- Low SG pressure <4.9 MPa
- Low core flow 2 of 4 MCP's tripped and power >75%
- High SG pressure PSG>7.84 MPa
- Low SG level LSG<0.65 M
- High Rx pressure PRX>17.6 MPa
- Low pressurizer level LPRZ<4.6 M

The initial condition was set to BOL which indicates 100% core power, core fuel loading is beginning of core life for equilibrium cycle and equilibrium XE and Sm. The reactor was operated in rated power condition for about 780 seconds in case of LBLOCA with LOOP and for about 838 seconds in case of LBLOCA with LOACP. Full rupture Dnom=850 mm (100%) of Cold Leg has been applied between Reactor and RCP. All 10KV normal power supply system from power grid have been disconnected for LB LOCA with LOOP scenario and 10KV emergency power supply has been disconnected for LB LOCA with LOACP scenario. The sequence of input command for the simulator has been described below.

| **Procedures that were followed to simulate LB LOCA with LOOP** | **Procedures that were followed to simulate LB LOCA with LOACP** |
|---|---|
| Simulator was started | Simulator was started |
| Initial condition BOL (IC-001) was applied for 100% core power | Initial condition BOL (IC-001) was applied for 100% core power |
| Normal rated operation for 780 seconds | Normal rated operation for 838 seconds |
| Malfunction was initiated by full rupture of cold leg (Dnom=850mm) between Reactor and RCP with all 10KV normal power supply disconnected from the grid. | Malfunction was initiated by full rupture of cold leg (Dnom=850mm) between Reactor and RCP with all 10KV normal and emergency power supply disconnected from the grid and Diesel Generator. |
| Observation of the transient data trend for 1020 seconds | Observation of the transient data trend for 962 seconds |
| Saved the data as '.csv' file | Saved the data as 'csv' file |
| End of simulation | End of simulation |

**Table 2.3:** The procedures to simulate the malfunction.

At the time of Simulation initiation, LBLOCA with LOACP had to be initiated manually. As a result, the scenario actuation time was not the same i.e. 780 second for LBLOCA with LOOP and 838 second for LBLOCA with LOACP. So, the time should be considered with respect to the accident initiation time. For both the cases, the reactor trip occurred followed by the turbine trip within 2 seconds. All the control rods were fully inserted to safely shutdown the reactor and to make the reactor subcritical.

## Result and Analysis

The numerical analysis for LBLOOCA with loss of offsite power (LOOP) and loss of AC power (LOACP) have been performed through VVER-1200 simulator. The reactor trip occurred due to loss of coolant. As a result, the power decreased rapidly and reached 1.7% from 100% within 3 seconds in case of LBLOCA with LOOP and from 100% to 2.7% for LBLOCA with LOACP. The power did not reach 0 immediately because of decay heat from the fission fragment. Both the scenario is shown in Figure 2.1a and 2.1b respectively.

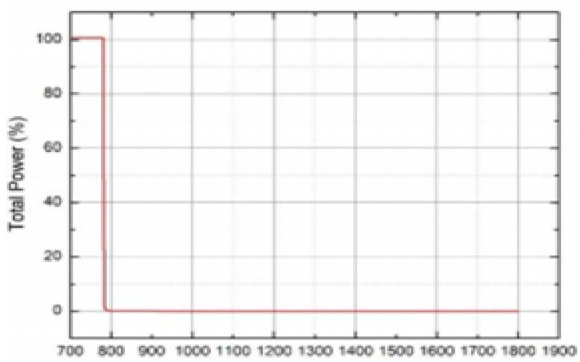


**Figure 2.1a:** Total power (%) for LOCA with LOOP.

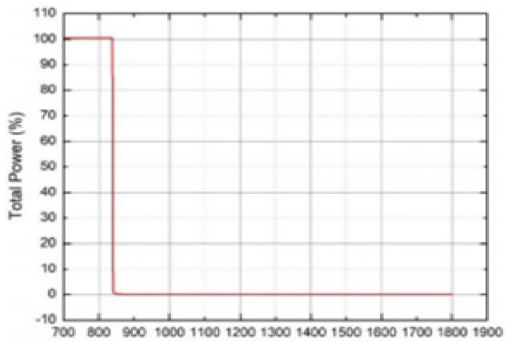


**Figure 2.1b:** Total power (%) for LOCA with LOACP.

As coolant lost from the primary circuit and the proportional heaters were not working, the pressure in the pressurizer (PRZ) decreased rapidly from 15.85 MPa to about 1 MPa within 18 seconds for LBLOCA with LOOP and within 17 seconds for LBLOCA with LOACP shown in Figures 2.2a and 2.2b correspondingly. The water level in the pressurizer also decreased abruptly and reached a steady level of about 3.2 m after 100 seconds for both the cases. This phenomenoa are shown in Figure 2.3a and 2.3b respectively.

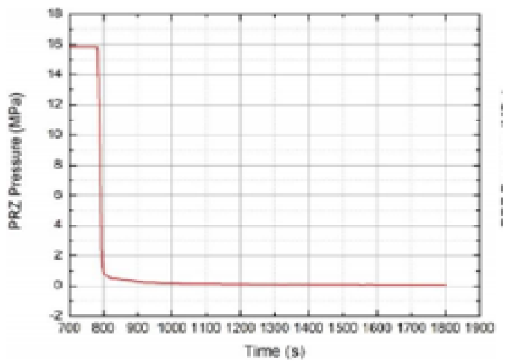


**Figure 2.2a:** PZR Pressure(MPA) for LOCA with LOOP.

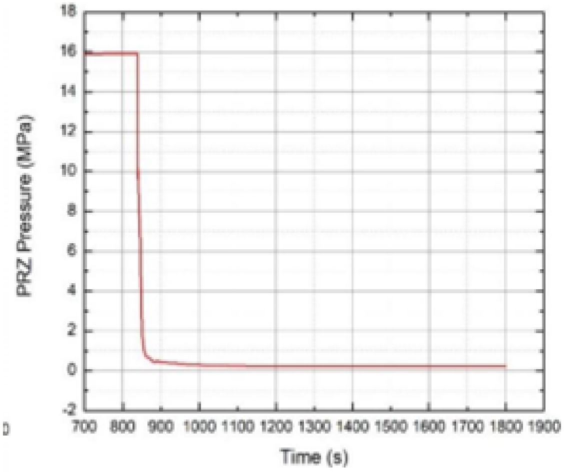


**Figure2.2b:** PZR Pressure (MPA) for LOCA with LOACP.

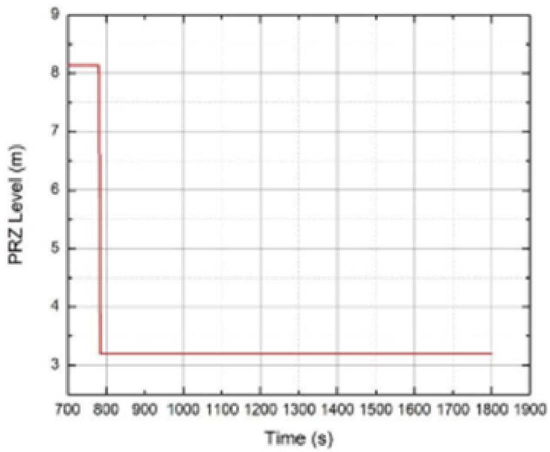


**Figure2.3a:** Water level (m) in PRZ for LOCA with LOOP.

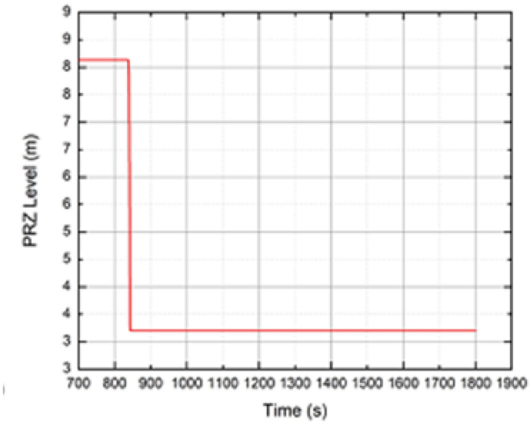


**Figure 2.3b:** Water level (m) in PRZ for LOCA with LOACP.

Due to trip of Reactor Coolant Pump (RCP) and Feed water pump (FWP), Steam generator pressures should increase following the immediate cessation of steam flow. The pressure is expected to increase until the steam generator atmospheric dump valves actuate. As steam is released from the steam generators, steam generator pressure should decrease and will continue decrease after Reactor trip. These phenomena are shown in Figure 2.4a and 2.4b respectively.

Steam generator level decreased initially due to shrink when the turbine trips. It has been seen that, as the backup diesel generators were working for the case of LBLOCA with LOOP, auxiliary feed water was supplied to the SG, and the water level increased. On the contrary, for LBLOCA with LOACP, diesel generators were off. As a result, there was no feed water supply to the SG, and SG water level decreased. These significant phenomena are shown in Figure 2.5a and 2.5b respectively.

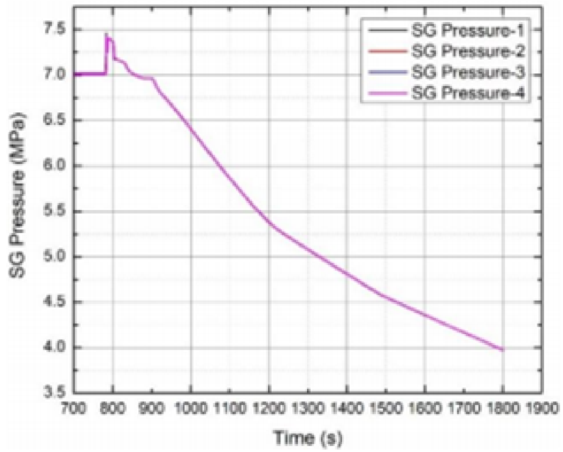


**Figure 2.4a:** SG pressure (MPA) for LOCA with LOOP.

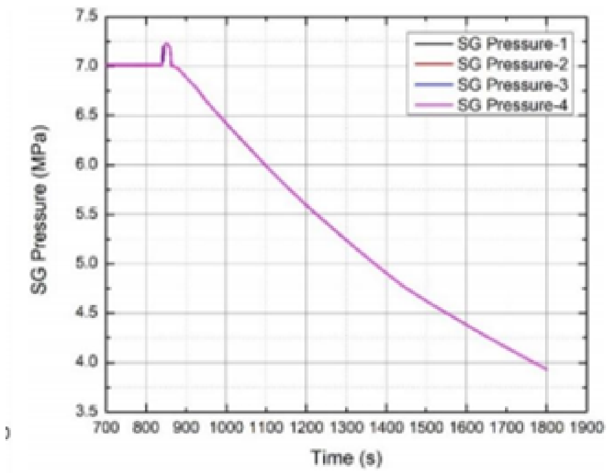


**Figure2.4b**: SG pressure (MPA) for LOCA with LOACP.

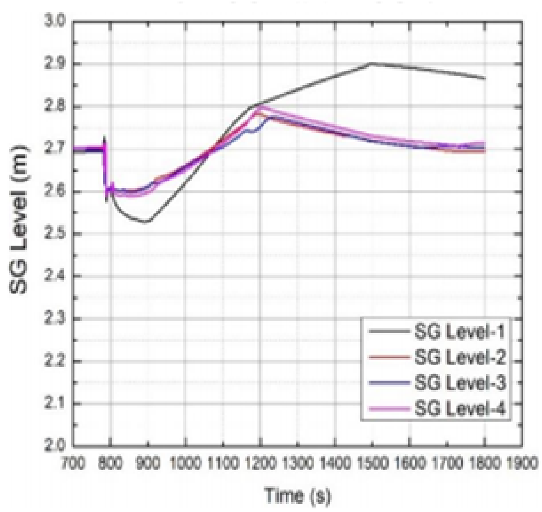


**Figure 2.5a:** SG water Level (m) for LBLOCA with LOOP.

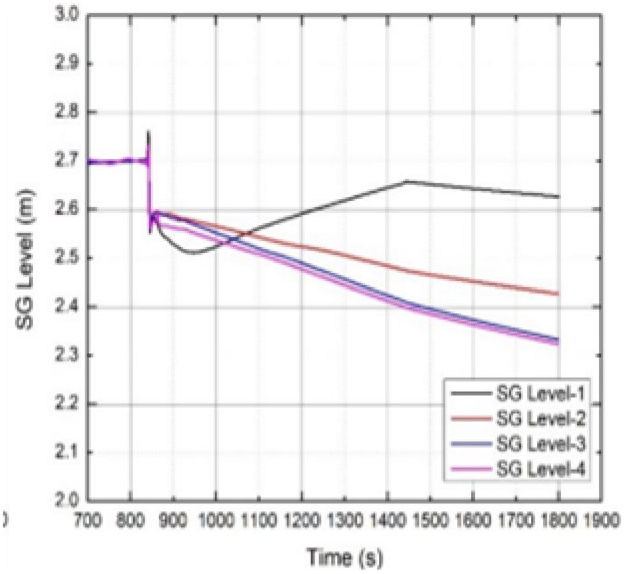


**Figure2.5b:** SG water level (m) for LOCA with LOACP.

Due to the reactor trip the temperature of the hot leg and cold leg are decreased. After 900seconds of the scenario actuation, the temperature reached about 104 °C in the hot leg and about 69 °C in the cold leg for LBLOCA with LOOP indicating in Figure 2.6a. With the similar time length, the temperature reached about 135 °C in the hot leg and about 125 °C in the cold leg for LBLOCA with LOCAP indicating in Figure 2.6b. The temperature in the hot and cold leg in the case of LBLOCA with LOACP is comparatively higher than that of LBLOCA with LOOP because the backup diesel generators provided coolant to the core using the emergency cooldown system in the scenario of LOCA with LOOP.

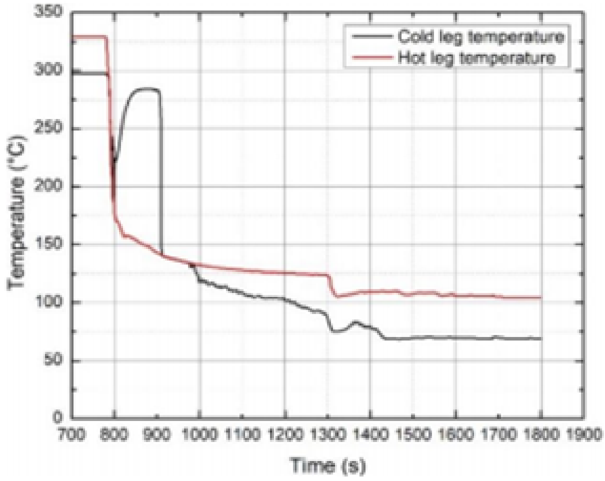


**Figure 2.6a:** Hot Leg and Cold Leg Temperature (°C) for LOCA with LOOP.

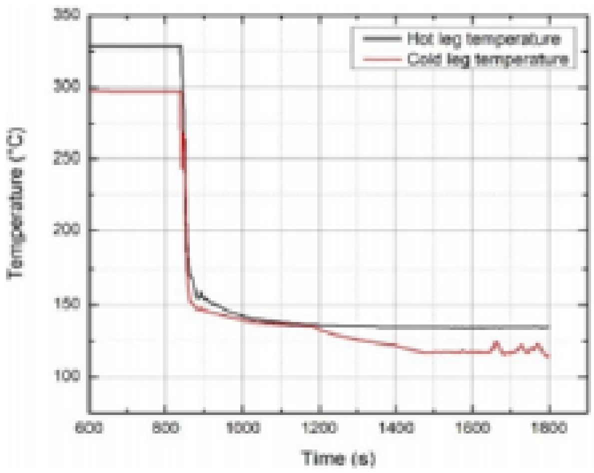


**Figure 2.6b:** Hot Leg and cold leg temperature (°C) for LOCA with LOACP.

Hydro accumulators are the passive component of ECCS which provides borated water to the core having a concentration of 16g/L, starts operating after the pressure drops below a prescribed level. The Water flow from Hydro accumulator through core is shown in Figure 2.7a and 2.7b. During LOCA, the coolant lost from the core resulting pressure decrease. The HA-1 started operating after the pressure dropped below 5.9MPa and had a maximum flow of about 1835 kg/s in the case of LBLOCA with LOOP and 1830 kg/s in the case of LOCA with LOACP. The HA-1 started within 7seconds and operated for about 194 seconds and 168 seconds respectively for both cases. The actuation of HA-2 occurred when the pressure dropped below 1.5MPa and had a maximum flow of 32kg/s and 40kg/s respectively for LBLOCA with LOOP and LB LOCA with LOACP. The HA-2 would continue supplying coolant to the core for about 24 hours respectively after the accident scenario.

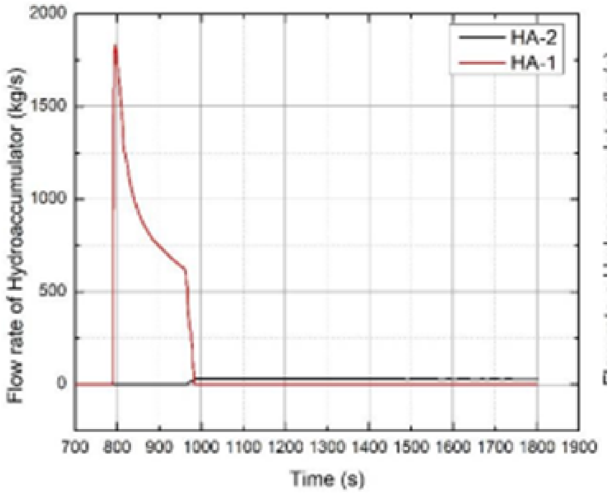


**Figure 2.7a**: Flow rate of hydro accumulator (kg/s) for LOCA with LOOP.

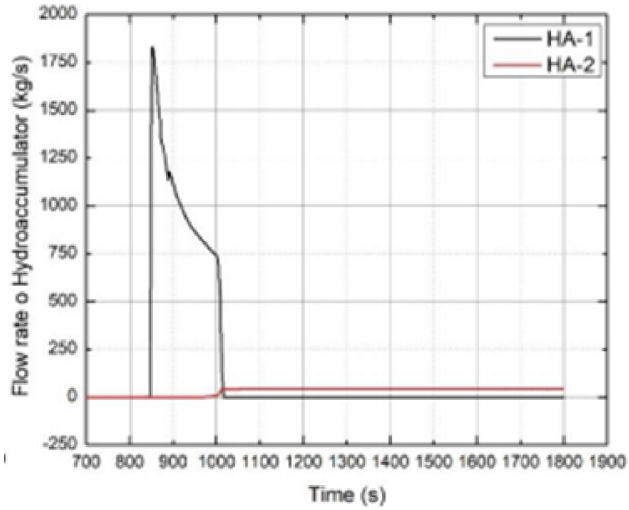


**Figure 2.7b:** Flow rate of hydro accumulator (kg/s) for LOCA with LOACP.

In normal conditions, the containment was kept at a negative pressure typically of -0.24kPa. The leaked coolant was contained in the containment which causes a rise in containment pressure. The pressure increased and reached a steady value of 40kPa for both cases.Initially, the containment temperature was 36 °C. After the accident, as hot water entered the containment building, the temperature increased to a maximum of 150 °C, then decreased because of the actuation of backup power (diesel generators), the Hydro-Accumulator (HA), Emergency core cooling system and Passive Heat Removal System in case of LOCA with LOOP scenario. On the other hand, temperature decreased slowly after reaching 150 °C due to absence of diesel generators and ECCS active part, only HA-1, HA-2 and Passive Heat Removal System are working in LOCA with LOACP scenario Figure 2.8a and 2.8b.

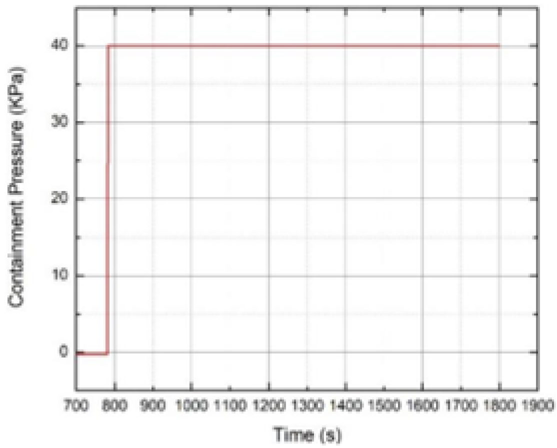


**Figure 2.8a:** Containment Pressure (kPa) for LOCA with LOOP.

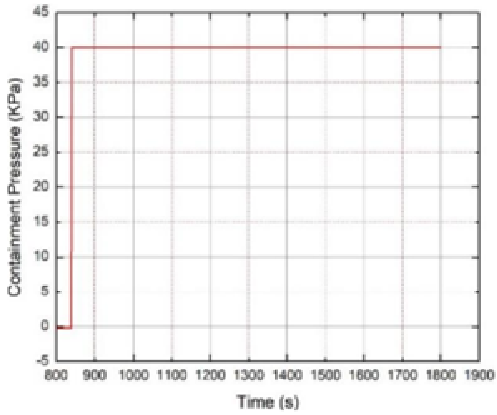


**Figure 2.8b:** Containment pressure (kPa) for LOCA with LOACP.

The Overall experimental simulation process is given through chronological sequence bellow Table 3.1 and 3.2.

| Time (sec) | Events | Interlocking, actuation of setpoints or other causes |
|---|---|---|
| 0 | Break in the primary pipeline. Trip of RCPS and other active equipment. | Initiating event<br>Loss of offsite power |
| 1 | Beginning of control rod movement. Containment pressure > 20 kPa Containment isolation. | Reactor scram action. Leaked hot water entered the containment. |
| 7 | Actuation of HA-1 | Reactor pressure reached 5.9 MPa |
| 10 | Hot leg temperature <260 °C | Coolant flow from HA-1 |
| 28 | Hot leg temperature < 150 °C | Coolant flow from HA-1 |
| 99 | Actuation of PHRS | Loss of offsite power |
| 136 | Actuation of HA-2 | Reactor pressure reached 1.5 MPa |
| 178 | End of HA-1 operation | Emptying of hydro accumulator tanks |
| 1020 | End of calculation | |

**Table 3.1:** Chronological sequence of events for the case of LBLOCA with LOOP.

| | | |
|---|---|---|
| 0 | Break in the primary pipeline concurrent with loss of all AC power. Trip of RCPS and other equipment. | Initiating event Loss of all AC (offsite and onsite) power. |
| 1 | Beginning of control rod movement. Containment pressure > 20 kPa Containment isolation. | Reactor scram action. Leaked hot water entered the containment. |
| 7 | Actuation of HA-1 | Reactor pressure reached 5.9MPa |
| 10 | Hot leg temperature <260 °C | Coolant flow from HA-1 |
| 28 | Actuation of PHRS | Loss of offsite power |
| 99 | Hot leg temperature < 150 °C | Coolant flow from HA-1 |
| 136 | Actuation of HA-2 | Reactor pressure reached 1.5MPa |
| 178 | End of HA-1 operation | Emptying of hydro accumulator tanks |
| 1020 | End of calculation | |

**Table 3.2:** Chronological sequence of events for the case of LB LOCA with LOACP.

## Conclusion

In above observation the total power has been observed to decrease more abruptly for LB LOCA with LOOP compared to LB LOCA with LOACP considering the severity of the later event. In comparison, the hydro accumulator first stage (HA-1) actuated at the same time in both cases but operated longer in the case of LB LOCA with LOOP due to actuation of the ECCS active part and SG emergency cool down system. On the other hand, the hydro accumulator 2nd stage (HA-2) started operating earlier for LB LOCA with LOACP and core cooling was entirely done by the hydro accumulators. The hydro accumulator results in the case of LB LOCA with LOACP were found to be in good agreement with MELCORE 1.8.6 code generated results [17]. The containment temperature reduced safely, and the containment pressure did not cross the safety margin. The pressurizer pressure, steam generator pressure did not exceed the safety margin prescribed by International Atomic Energy Agency (IAEA) safety reports.It has been observed that LB LOCA with LOACP is more severe compared to LB LOCA with LOOP. Based on the results it is evident that the VVER-1200 reactor will be safe during the major design extension scenario of LBLOCA with LOOP or LBLOCA with LOACP.

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