

Analysis of Nuclear Reactor Parameters in the VVER-1200 during a Loss of Coolant Accident and Steam Generator Tube Rupture with Emergency Core Cooling System Failure

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ABSTRACT

Nuclear energy is widely recognized as a clean energy source and has gained popularity for its role in electricity production. It currently accounts for around 10% of global electricity production, with nuclear reactors being a key element in the operation of nuclear power plants. Numerous reactor types have been developed over the years but the VVER-1200, a Generation III+ PWR, is implemented worldwide due to its advanced safety features. In this work, PCTTRAN, a thermal-hydraulic modeling software has been utilized to simulate fault circumstances on the VVER-1200 reactor with or without the emergency core cooling system (ECCS) in operation. As this software encompasses several built-in malfunctions, steam generator tube rupture (SGTR) with 100 cm² break at B, and break loss of coolant accident (LOCA) at cold leg with 856 cm² failure have been introduced into the software to analyze the impact of the faults on the nuclear plant parameters such as pressure, temperature, reactivity, water level, and so on. Following the SGTR and LOCA, pressure, temperature, power, and coolant flow dropped drastically within the first 200 seconds after the reactor got tripped at 13 secs. With ECCS in operation, the accumulator valve provides water from the emergency water storage tank (EWST), stabilizing power, fuel temperature, reactor coolant flow, and pressure, thereby preventing core damage. However, ECCS failure results in core exposure, fuel melting, radioactive leakage, and core collapse from molten core concrete interaction.

Keywords: Renewable energy, Nuclear energy, VVER 1200 reactor, PCTTRAN, Loss of coolant accident (LOCA)



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1. Introduction

All the countries in the world along with Bangladesh are experiencing severe energy crises. Though Bangladesh has several natural energy resources, it will eventually run out as most of the sources are only being used for generating electrical power. But this massive energy crisis could be reduced using nuclear energy. Though it has been used for destructive purposes, it can be utilized to generate electricity in the modern era as a tremendous quantity of power can be generated if the chain reaction is regulated in a controlled way in the reactor. A nuclear reactor is a structure in which nuclear chain reactions get initiated and controlled in a specific way to generate energy in nuclear power plants (NPPs) without getting interrupted by any surrounding means [1, 2]. There were 437 nuclear reactors worldwide, which were in an operable state as of December 31, 2021, by the International Atomic Energy Agency (IAEA), with a total capacity of 389.5 GW(e). As of today, this number has increased alongside nuclear power capacity, with 20.7 GW(e) added capacity over the previous decade [3]. Different types of reactors have been introduced in recent years such as Pressurized-Water Reactors (PWR), Boiling-Water Reactor (BWR), Advanced Gas-cooled Reactors (AGR), Elevated Temperature Gas Cooled Reactors (HTGCR), Heavy Water Nuclear Reactors (HWR) and more researches are ongoing. Among them, VVER-1200 is the most sophisticated, PWR-based nuclear reactor in terms of design and efficiency, where water serves as both coolant and moderator. VVER is an abbreviation for water-water energy reactor, which was developed by the Russian Federation's Gidropress Podolsk Company. In 2009,

Bangladesh and Russia came to an agreement regarding the construction of an NPP called Rooppur Nuclear Power Plant (RNPP) at Ishwardi in the Pabna district, with Russian State Atomic Corporation 'Rosatom' in charge. This is the world's largest power plant project, led by Rosatom and co-directed by the Bangladesh Atomic Energy Commission (BAEC) and the Atomic Energy Commission of India (AECI) [1, 2].

NPPs are designed to minimize the risk of accidents and protect people through safety measures and risk management. These plants are equipped with multiple protective systems to handle abnormal conditions and have robust emergency protocols to ensure safe shutdowns. In extreme cases, however, core damage and containment breaches may still occur, leading to potential radioactive release. To mitigate such risks, every plant maintains severe accident management plans, that are frequently reviewed and improved to halt core damage, preserve containment, and minimize radioactive emissions [4]. Moreover, a leak in the coolant system can lead to a substantial coolant loss from the Reactor Cooling System (RCS), resulting in either a Design Basis Accident (DBA) or a Beyond Design Basis Accident (BDBA), depending on the severity of the leak. DBAs are predefined accident scenarios that nuclear plants must be designed to handle without compromising the structures and systems essential for ensuring public safety. These accidents test the performance of reactor structures and components during unusual operations. On the other hand, BDBAs are rare events with a low probability of occurrence, which might or might not lead to core degradation. DBAs cover a range of potential failures, such as coolant losses and reactivity-initiated accidents, with specific thresholds for

cladding temperature and radiation exposure [5]. Because of this, plant-specific models have been established in nuclear power plants and universities worldwide for practical implementation in training, inspection, and safety assessment, and to lower the chances of nuclear accidents. The IAEA has chosen PCTTRAN, the most effective personal computer simulator for every kind of light water nuclear reactor including PWR, BWR, Westinghouse AP1000, etc., as a training tool for the annual IAEA Simulators Workshop. Windows-based graphical user interfaces enable users to engage with simulation software by directly manipulating graphical components. Because of its user-friendliness, it can be run faster than real-time. The nuclear facilities were developed and designed in such a way that they could survive both DBAs and BDBAs. Microsimulation technology's most recent upgrade to PCTTRAN includes several faults, which involve core melting, steam generator tube rupture, LOCA, station blackout (SBO), loss of feedwater flow, and corium-concrete interaction. These faults are considered as an initiating event for the DBAs and their potential impacts on different parameters such as feed water flow, reactivity, leak mass flow rate, pressure, turbine load, steam flow, etc. can be investigated through transient behavior analysis. Therefore, this program contains various aspects making it appropriate for the use of an assessment tool for emergency responses [2, 6].

LOCA is frequently encountered as a DBA in reactor design. It involves a situation where an inlet or outlet pipe within the RCS is fractured or partially broken, resulting in coolant leaking from the rupture. This type of accident can be caused by various conditions or operational behaviors of the core and associated systems [1]. Furthermore, steam generators in pressurized water reactors typically have 3,000 and 16,000 tubes that allow heat transfer from primary water to secondary water, producing steam to power turbines. After passing through the condenser, steam is returned to the steam generator via feedwater pumps. Over time, these tubes may suffer from cracks or ruptures, indicating an SGTR accident of a nuclear reactor that may lead to radioactive leaks. Therefore, maintaining these tubes is essential to prevent such issues. During a LOCA accident, the ECCS is triggered automatically to maintain water coverage and cooling for the reactor core [7]. But, in case any natural events like floods, tsunamis, and earthquakes could affect the performance of the ECCS and lead to the failure of diesel engines responsible for emergency backup power. While the probability of this type of hypothetical accident is very low, the geographic and socio-economic factors could still pose a disaster risk to the NPP [1].

Despite a limited amount of research on VVER-1200 reactor-related topics, increasing demand has motivated numerous researchers to explore nuclear accidents, their causes, and safety responses. The research conducted by Ibrahim et al. simulated several transient and accident conditions in nuclear power plants using PCTTRAN. Their LOCA simulation showed a fast decline in coolant pressure at 21.8 kN/m²/sec, triggering the RPS (reactor protection system) to shut down the reactor. The sudden shutdown of the turbine resulted in a significant drop in plant power at 14.3 MWe/sec, causing operational downtime [8]. On the other hand, Fyza et al. explored the performances of thermal as well as hydraulic response of the VVER-1200 at the time of the LOCA accident paired with a loss of offsite power. The power loss was initialized after 500 seconds in both simulations, leading to a decrease in RCS pressure transitioned from 155 bars down to less than 5 bars, effectively reducing leakage. After 23.5

seconds, the reactor core's thermal power along with nuclear flux, and turbine load experienced a significant drop, while fuel temperature decreased below 200 °C between 150 and 200 seconds after the malfunction [9]. Furthermore, Nath et al. evaluated the thermal-hydraulic aspects of a VVER-1200 reactor under LOCA break conditions to assess accident impacts. The study found that pressure, temperature, and power decreased rapidly after LOCA and SBO events, leading to a reactor trip. The core water level dropped to 4 meters after 65 seconds, triggering the Hydro-Accumulator to prevent core damage. Reactor thermal power, turbine load, and nuclear flux power fell within 18 seconds of the accident. The research concluded that while ECCS can mitigate LOCA with SBO, its failure results in core uncovering, fuel melting, radioactive release, and core collapse through molten core-concrete interaction [1].

This research explores the issue of steam generator tube rupture by simulating it in SG-B with LOCA scenario either in the presence or failure of the ECCS system during the transient phase. This study examined these faults to analyze how the reactor manages plant variables like reactor pressure, temperature, core water level, core thermal power, and triggers emergency systems to minimize known hazards.

2. Methodology

To initialize the operation of the reactor, certain initial values are required. Several characteristics of the reactor such as pressure, thermal power, temperature, and coolant flow can be described by initial conditions of the system before the occurrence of the accident. All these factors can be assessed on the plant. As PCTTRAN software has been used in this research work, it has some built-in initial steady-state conditions. In this case, #1 was used as the initial condition in the simulation process. The following Table 1 has the values of the initial conditions which was used during the research work.

After selecting the initial condition, 5 seconds was considered to run the simulator before introducing malfunctions in the simulation process to stabilize the reactor. After that, LOCA in the cold leg with 856% area (13inch diameter) break along with tube rupture in steam generator B 100% area failure were introduced from the malfunctions list in this research work, simulating in presence or failure of the ECCS system.

Table 1 Values of initial conditions used for simulation.

Parameter	Value
IC number	1
Power	100
Reactor core pressure	155
Set point for steam generator isolation, T_{avg}	306.9
Steam generator pressure	70
TimeInLife	BOC

After introducing all the malfunctions to the simulator, it kept running for 300 seconds. After 300 seconds, the state of the plant in the simulator was saved as an initial condition and then again it was run for 300 seconds. Each time the simulation was run for 300 seconds and it was done 5 times to observe the condition of the plant during accidents till 1500 seconds.

3. Result and Discussion

Two scenarios were explored in this computational analysis. The first was a design basis LOCA with tube rupture throughout an active operation of the ECCS, and the second involved a beyond-design basis LOCA and tube rupture with neither the ECCS nor the passive cooling system functioning. The discussion was centered around the plots generated for various reactor parameters after the analysis.

3.1 Analysis of LOCA at cold leg and tube rupture at SG-B with ECCS

This numerical analysis considers an 856 cm² (13-inch diameter) line break in the cold leg stream. A malfunction was manually introduced with an 856 cm² failure fraction at the cold leg, along with a tube rupture at B (SG-B), 5 seconds after starting the simulator at 100% power. Fig.1 illustrates the PCTRAN interface once the 100-second simulation has been carried out.

Fig.2(a) illustrates the coolant flow through the reactor cooling system (RCS) leak which indicates that, after occurring the accident, a significant amount of reactor coolant is lost from the system, which in turn depressurizes the system and the pressure of the RCS starts to fall from 162 bar that is further depicted in Figure 2(b). Due to the accident, there is a rapid drop in the mass flow rate within the first 200 secs. Eventually, the mass flow rate stabilizes, showing the system's near-equilibrium state at lower internal pressure over a prolonged time.

During the initial phase of a LOCA with tube rupture, the reactor loses coolant faster than the pressurizer can compensate for, leading to a fast decline from 11.2 m in the water level which is shown in Figure 3(a). The high-pressure injection system (HPIS), eventually begins to compensate for this loss, but there is a period where the pressurizer empties because the coolant is escaping from the system faster than it can be replenished. After 200 secs, the pressurizer starts to refill on its own as the ECCS injects coolant into the system, restoring pressure and water levels. When the reactor gets tripped at 13 secs, control rods automatically get inserted into the core and the core water level starts to decrease within the reactor. Upon actuation of the accumulator valve at 59 bar, ECCS starts to provide borated water from EWST directly into the reactor core, ensuring that the fuel rods remain submerged and cooled so that overheating and potential core damage can be prevented.

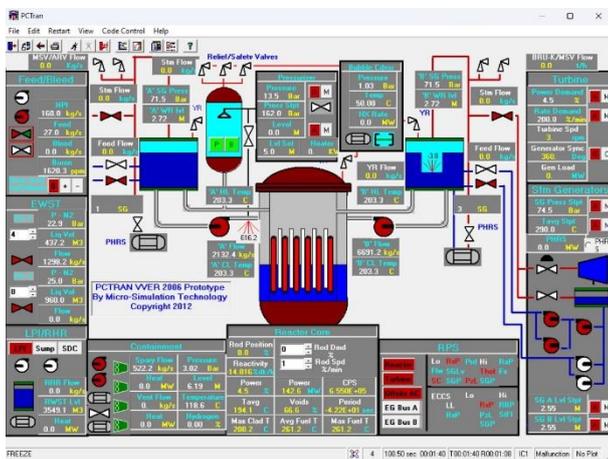
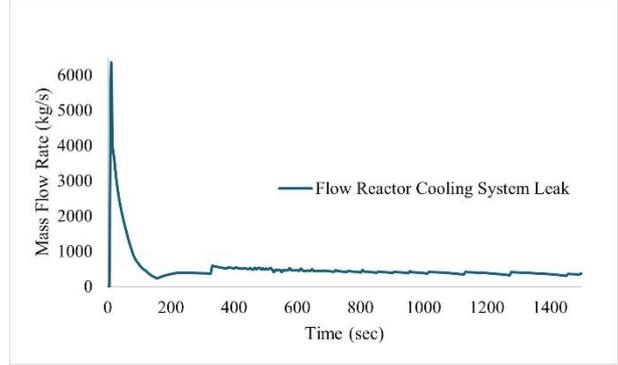
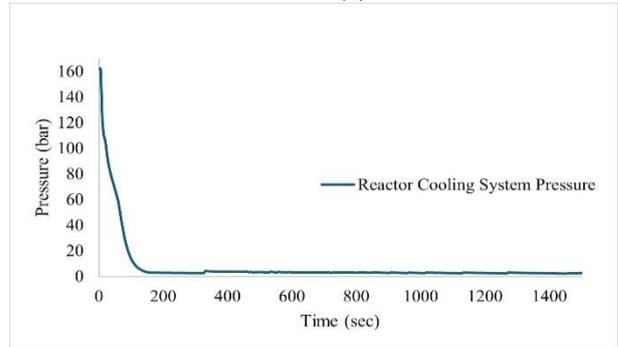


Fig.1 PCTRAN interface after conducting the simulation for 100 secs.



(a)



(b)

Fig.2 Reactor Cooling System (a) Leak Flow Rate (kg/s) vs Time (secs) (b) Pressure (bar).vs Time (secs) for LOCA at cold leg and SGTR at B with ECCS.

When the ECCS starts to compensate for the coolant loss, there is an increase in the core water level at 90 secs. After that, the core water level reaches a stable condition, indicating effectively maintaining water level in the core. During the accident, leaked coolant from the RCS collects in the sump, increasing the gradual accumulation of water in the reactor building which is also illustrated in Fig.3(a). Furthermore, the rapid decrease in both fuel and cladding temperature signifies the reactor's ECCS effectiveness in reducing heat generation during accidents, as seen in Fig.3(b). In addition, Fig.3(c) presents a fast decline in core thermal power, turbine load, and nuclear flux power within the first 100 seconds, indicating the reactor's immediate safety systems response to LOCA and SGTR. Afterward, the values settle close to zero, showcasing the reactor's secure shutdown state throughout the 1500-secs observation.

3.2 Analysis of LOCA at cold leg and tube rupture at SG-B without ECCS

As VVER-1200's ECCS is equipped with HPIS, LPIS (Low-Pressure Injection System), passive heat removal systems, hydro accumulator, accumulator valve, feedwater systems, reactor spray pumps, and RWST (Refueling Water Storage Tank), all these systems work together to handle emergencies, from DBAs to BDBAs. Upon detecting an accident, the ECCS activates automatically, ensuring the core remains submerged and cooled to prevent serious harm to the reactor, environment, and staff. However, a failure in the ECCS could result in a severe accident, leading to core meltdown and fuel degradation [5].

In this analysis, an 856 cm² LOCA at the cold leg and SGTR at B with the failure of all ECCS equipment, is simulated. Even though this is a hypothetical situation, the Three Mile Island Unit 2 (TMI-2) accident driven by design flaws and mechanical failures, classified as a level 5 accident,

became a significant learning point for the nuclear industry. Moreover, the Fukushima Daiichi accident where all emergency systems failed after an earthquake and tsunami, leading to a level 7 disaster, emphasizes the need to analyze the possible consequences of these similar events.

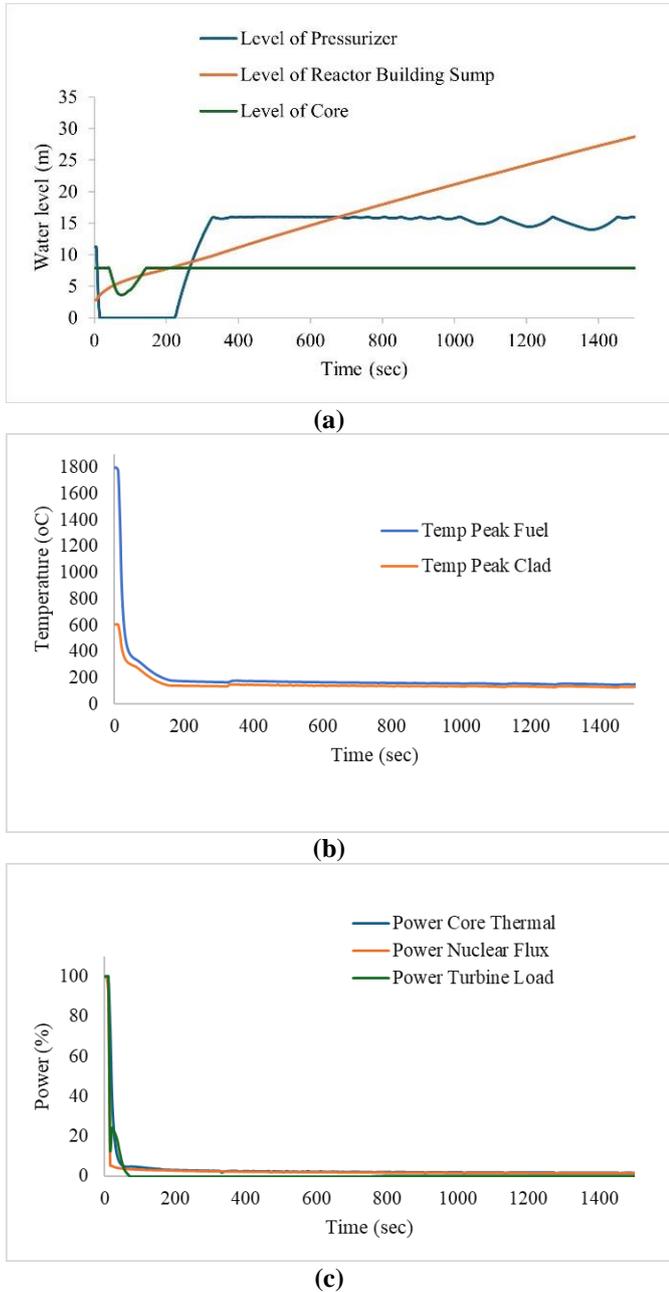


Fig.3 (a) Water level in different sections (m) vs Time (secs) (b) Temperature peak fuel and clad (°C) vs Time (secs) (c) Power of core thermal, nuclear flux and turbine load (%) vs Time (secs) for LOCA at cold leg and SGTR at B with ECCS.

Here, Fig.4(a) represents the reactor state with an emptied pressurizer at 22 secs in occurrence with the failure of ECCS accompanied by LOCA at cold leg and SGTR at B. After that, Fig.4(b) and Fig.4(c) show the core uncover status where the coolant level drops below the top of the reactor core, exposing the core partially or fully, leading to the core melting process. Furthermore, Fig.4(d) illustrates a severe condition where the vessel is being penetrated due to an extreme temperature rise within the reactor core and the failure to restore cooling at the same time.

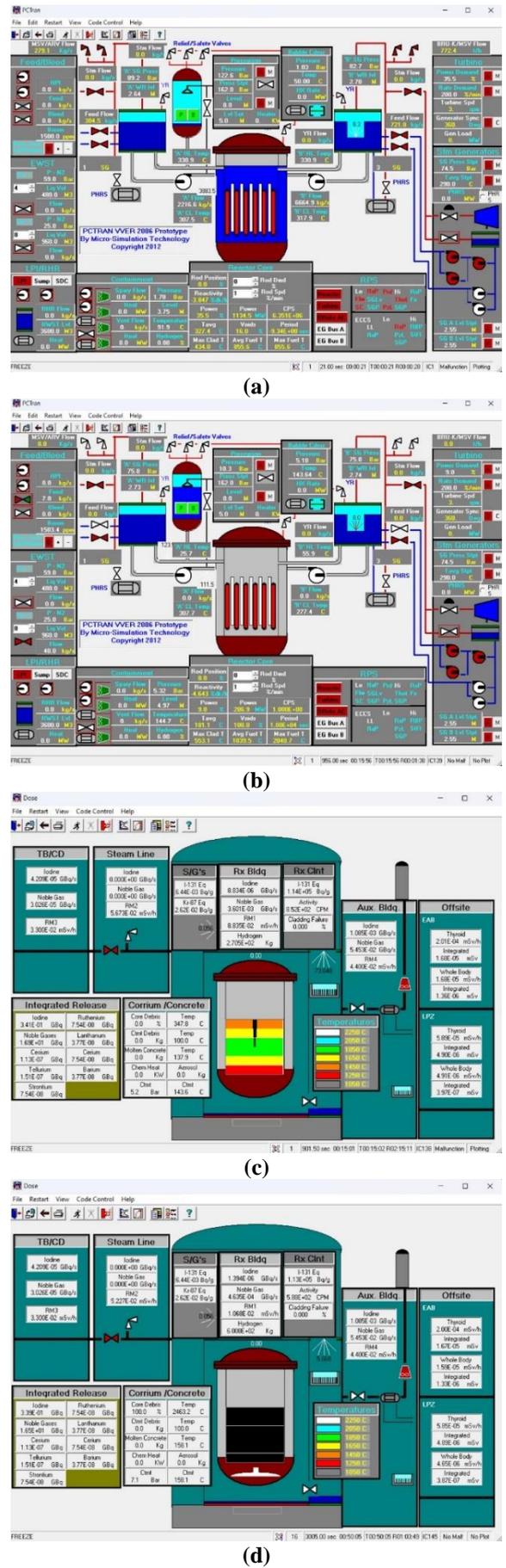
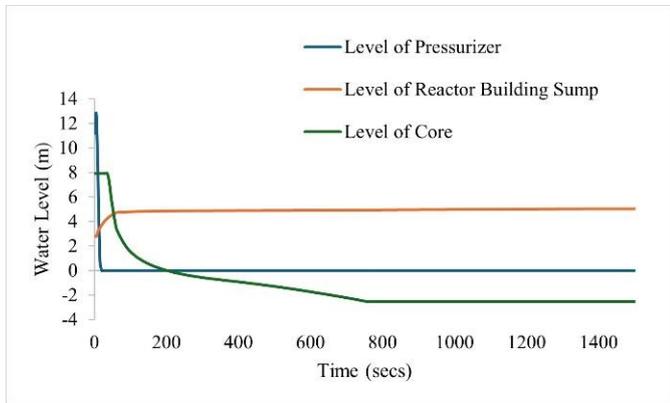
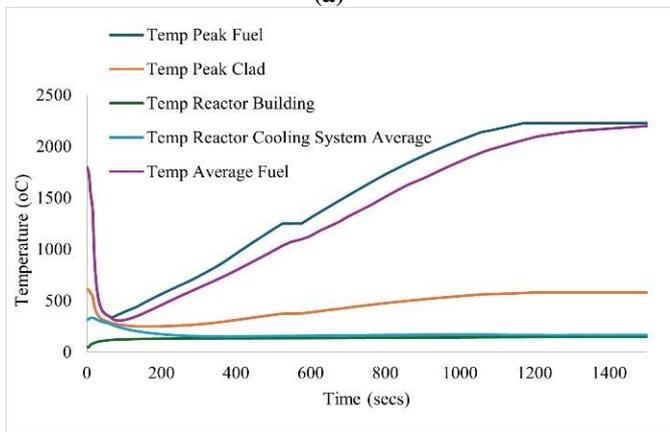


Fig.4 (a) Reactor state at 21 secs (b) Core uncover (c) Core melting process in dose mimic (d) Vessel penetration for LOCA at cold leg and SGTR at B without ECCS.

Without enabling ECCS in a LOCA and SGTR accident, due to the rapid coolant loss, followed by depressurization, triggers reactor trip within 16 seconds, with all control rods fully inserted into the core. In the meantime, the core water level starts to drop drastically with an increase in sump water level until recirculation ceases with no water left to transport the poison, as seen in Fig.5(a). Fig.5(b) shows the temperature of various components after a LOCA in the cold leg and SGTR, along with ECCS failure in a VVER-1200 reactor. The fuel temperature rises significantly, indicating that the core is not being adequately cooled after the coolant loss. As the temperature surpasses 2000°C, severe overheating of the core occurs. Moreover, the cladding temperature increases at a slower rate due to its lower thermal properties, which means it takes longer to heat up. The reactor building temperature remains comparatively low compared to the fuel and cladding, implying that heat transfer from the core to the containment is minimal due to containment isolation or failure to extract heat from the primary circuit.



(a)



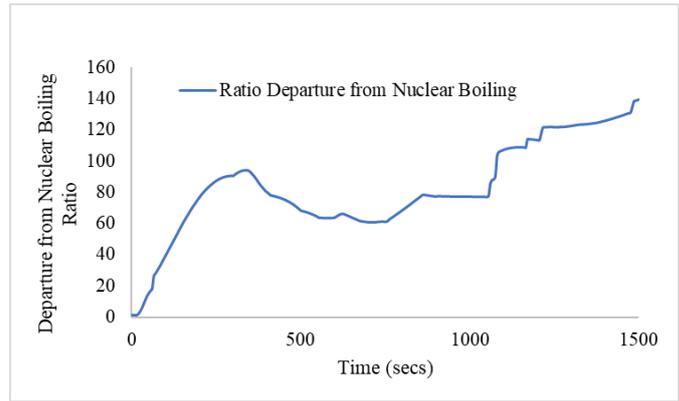
(b)

Fig.5 Different sections (a) water level (m) vs Time (secs) (b) Temperature (°C) vs Time (secs) for LOCA at cold leg and SGTR at B without ECCS.

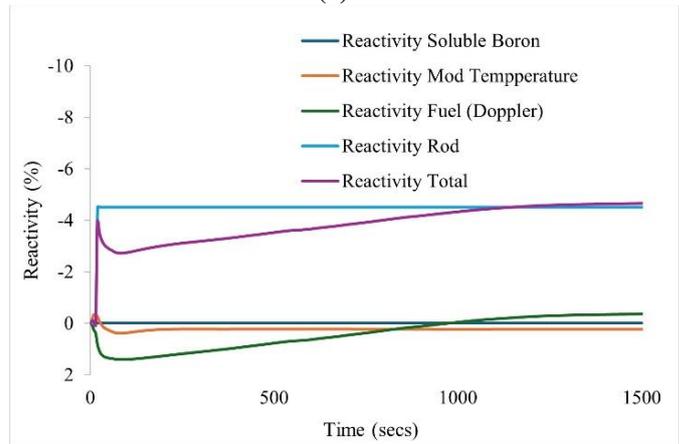
DNBR (Departure from Nucleate Boiling Ratio) is defined as the ratio of critical heat flux to the local heat flux at a given point in a nuclear plant. Fig.6(a) presents the gradual increase of DNBR up to 80 as the local heat flux remains below the critical limit until the complete loss of coolant.

Once the local heat flux surpasses the critical limit, the core materials start melting as the heat flux continues rising. The increase in fuel temperature inside the core causes a fractional reactivity change called fuel Doppler reactivity

and remains safe below 1 but an explosion leads to a rapid increase in reactivity across different sections, dispersing radioactive substances, as depicted in Fig. 6 (b).



(a)



(b)

Fig.6 (a) Departure from nuclear boiling ratio vs Time (secs) (b) Reactivity of soluble boron, modifier temperature, fuel doppler, rod, and total (%) vs Time (secs) for LOCA at cold leg and SGTR at B without ECCS.

4. Conclusion

This research illustrates the accidental state of LOCA and SGTR at the same time with or without having ECCS in an NPP by analyzing various plant factors. According to this analysis, ECCS can mitigate the LOCA with SGTR by providing water directly into the core but the accident can turn into disaster if ECCS is somehow unavailable or malfunctioned. ECCS failures require a comprehensive approach to reactor safety that incorporates technical, operational, and procedural improvements. Technically, deploying advanced sensors to monitor ECCS parameters like pressure, flow rate, and temperature allows for early fault detection. Moreover, if possible, separating system components physically could help prevent simultaneous failures caused by external events such as fires or floods. On the other hand, operational measures include ongoing training programs, emergency drills, and the implementation of real-time monitoring systems. By adopting these measures, VVER-1200 operators can bolster reactor safety, reduce ECCS-related risks, and align with global nuclear safety norms. Since a fault can cause a nuclear power plant to trip, essential precautions should be carried out in time to mitigate accidents because a reactor trip can disrupt the whole electrical network. Any minor accident can quickly escalate into a major calamity leading to spreading radioactivity, if

not handled appropriately. In addition, a serious accident is not only deadly for humans but also for all other living beings in the surroundings of the plant.

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