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Holistic Approach to Safety and Operational Stability: Analyzing *VVER*-1200 Reactor Dynamics in SGTR and AC Power Loss

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Abstract

VVER-1200 (Water-Water Energetic Reactor) represents a significant advancement in nuclear power generation, emphasizing the continuous analysis and enhancement of safety systems for reliable operation. The proposed study focuses on simulating combined scenarios involving steam generator tube rupture (*SGTR*) and *AC* power loss using core algorithms and models within personal computer transient analyzer (*PCTRAN*). Reactor kinetic equations, thermal-hydraulic balance, and safety system models are discussed to elucidate their role in simulating *SGTR* and *AC* power loss. Safety criteria, boundaries and initial conditions are outlined to provide a comprehensive understanding of the simulation framework. The analysis delves into dynamic behavior of *VVER*-1200, placing emphasis on thermal-hydraulic implications, essential reactor parameters, and radiation monitoring to facilitate impact evaluation. Continuous monitoring and maintenance of safety systems are underscored to ensure stable core cooling, particularly during proposed transient conditions. Through meticulous analysis and comparison with established benchmarks, this study contributes to bolstering the safety and reliability of *VVER*-1200 reactors by identifying vulnerabilities, assessing mitigation strategies, and refining emergency response protocols. Practical implications of this study offer a crucial understanding of reactor behavior, safety system performance, and emergency response strategies, thereby improving safety, optimizing operational practices, and reducing risks in nuclear reactor accidents.

Keywords

VVER-1200, Nuclear Technology, Safety Systems, Operational Efficiency, Transient Events

1. Introduction

In the realm of nuclear power, safety is paramount due to potential catastrophic consequences of safety lapses, emphasizing the inherent risks associated with nuclear energy and the need to prioritize safety. *VVER*-1200 (water-water energetic reactor), a significant advancement in nuclear technology, is a testament to the evolution of nuclear power. Its advanced safety systems [1] and operational efficiency [2] make it a pivotal milestone in sustainable and secure energy generation. However, the quest for safety is ongoing [3], necessitating comprehensive analyses to identify potential vulnerabilities and devise effective mitigation strategies. Analysis of safety systems, operational parameters, and neutronics dynamics during transient events *i.e.* steam generator tube rupture (*SGTR*), *AC* power loss as well as combined event of the mentioned trainsets is critical in fortifying the safety measures of the reactor. *SGTR* incidents can impact the reactor coolant system [4], [5] necessitating a comprehensive analysis of safety systems and operational parameters evaluation to determine potential consequences and implement preemptive safeguards. The loss of *AC* power also requires thorough scrutiny to maintain safety functions [6], [7]. However, accurate modeling and analysis of transient events such as *SGTR* and *AC* power loss through simulation present challenges for nuclear power plant (*NPP*). These challenges include precise transient event parameterization, complex dynamic interaction modeling, identification of transient safety vulnerabilities, and effective decision-making plant operations. Hence, accurately simulating of the mentioned transient events is essential, as it must provide precise datasets to address parameter deviations from normal conditions, offer realistic feedback, and undergo continuous improvement and adaptation to remain relevant and effective.

The proposed study aims to conduct an in-depth analysis of the dynamic behavior of nuclear power plants with a specific emphasis on understanding the response of key components and safety systems during multifunctional transient events. By focusing on the intricate dynamics of *VVER*-1200 operations, the study seeks to understand the behavior of critical systems and components under varying transient conditions, ultimately contributing to the enhancement of safety protocols and operational resilience. By employing advanced simulation techniques, the study aims to provide valuable insights into plant performance and safety as well as radio-logical consequences, contributing to the enhancement of nuclear power plant resilience. The goal of the study is to assess the impact of transient events on *VVER*-1200 nuclear power plant and evaluate the effectiveness of safety measures in mitigating potential risks. By conducting comprehensive analyses of thermal-hydraulic parameters, reactor kinetics, safety system responses, radiation levels and releases in nuclear steam systems due to transient events to identify areas for improvement in plant design and operation, the proposed study aims to contribute to the development of enhanced accident management strategies and operator training protocols. Hence, primary focus of this study is to analyze the dynamic behavior of *VVER*-1200 under transient conditions, particularly during *SGTR*, *AC* power loss and combined scenarios with mentioned transients. The objectives of the study are to replicate transient events, conduct synergistic investigations of thermodynamic and neutronic aspects of nuclear reactor safety parameters, evaluate the effectiveness of safety systems, identify potential vulnerabilities in plant design and operation, and contribute to the body of knowledge on nuclear power plant dynamics under transient conditions. The proposed study enhances the safety and reliability of *VVER*-1200 by understanding and mitigating potential risks associated with transient events. By conducting rigorous analyses and simulations, the study aims to provide valuable contributions to plant behavior under adverse conditions and contribute to the development of advanced accident management strategies. The goal is to ensure the continued safe operation of nuclear power facilities and support the transition to clean and sustainable energy sources.

The proposed study examines the thermal-hydraulic complications of *SGTR* and *AC* power loss in *VVER*-1200 reactors, highlighting the potential release of radioactive material if not managed properly. The loss of *AC* power can compromise cooling systems and safety mechanisms, potentially leading to overheating and component failure [8] [9]. Factors contributing to *SGTR* with *AC* power loss include the dependency of critical safety systems on continuous power supply, potential component failure, and the need for effective backup power and emergency cooling mechanisms. To mitigate risk, robust backup power systems, emergency cooling measures, and redundant safety features are essential.

There has been a growing interest in studying behavior of reactors under various transient conditions using computational code and simulation [10]-[12] as well as the simulation of normal operating behavior such as coolant flow through subchannel in pressurized water reactor (*PWR*) [13].

Jintang and Chen examined the *SGTR* accidents in *PWRs* and the management strategies used in response, particularly in advanced *NPPs* [14]. *SGTR* accidents are significant due to their potential frequency and severe consequences involving radioactive material release. The authors emphasize the importance of maintaining coolant inventory to prevent overfilling of damaged *SGs* and subsequent release of radioactive materials into the environment. Traditional *NPPs* rely heavily on operator intervention to manage *SGTR* accidents, but advanced designs like passive pressurized water reactors (*PWRS*) aim to minimize this dependency through passive safety systems. *PPWR* incorporate Human Factors Engineering during design to ensure safe operational limits. Matejovic *et al.* investigated the detailed analysis of strategies to prevent and mitigate the consequences of a station blackout in nuclear power plants (*NPPs*), with a specific focus on the *WWER*-440 reactor design [15]. The study employs *RELAP*5/*Mod*3.2 code to perform support analyses for the development of Emergency Operating Procedures (*EOPs*) for these *NPPs*. The paper outlines the sequence of events following a station blackout, including reactor trip, loss of feedwater, and decay heat removal process through natural circulation. It emphasizes the importance of early recovery of electricity supply to prevent serious core damage and highlights the robustness of *WWER*-440 reactors compared to typical western Pressurized Water Reactors. Several strategies for managing a plant blackout are proposed, including passive primary and secondary side "bleed feed" methods and the use of special pipework with adapters for mobile fire engines. Hossen *et al.* examined Steam line break (*SLB*) accidents in *VVER*-12000 using *PCTRAN* simulator [16]. The study simulates five hypotheticals *SLB* accident scenarios, each lasting 300 seconds. Key parameters like reactor coolant system pressure, steam generator pressure, pressurizer liquid level, and break mass flow rate are analyzed. The study finds that smaller break sizes result in higher *RCS* pressure and temperature due to smaller coolant inventory loss. The study also discusses safety aspects, noting no significant changes in peak cladding temperature, peak fuel temperature, or radiation levels. A comprehensive examination of radiological source term behaviors during *SGTR* accidents, with a specific focus on the consequences of *AC* power loss, has been conducted by Esfandiari *et al.* [17]. The authors emphasize the importance of considering *SGTR* scenarios, even though their frequency might be lower compared to other accident sequences. The study uses *MELCOR* code to simulate a worst-case *SGTR* scenario in a Korean Optimized Power Reactor (*OPR*-1000) plant, considering factors such as reactor trip, safety system failure, and radioactive material release. Darnowski *et al.* conducted a study on the thermal-hydraulic behavior of *VVER*-1000 reactor under a hypothetical scenario of total loss of *AC* power [18]. The study uses *RELAP*5 code to analyze the thermal-hydraulic behavior of primary and secondary systems, including reactor core, circulation loops, steam generators, and safety systems. The analysis reveals a sequence of events following the loss of *AC* power, including reactor scram, turbine trip, loss of feedwater, and shutdown of main coolant pumps. A detailed analysis of a primary-to-secondary leak accident scenario at the Bushehr nuclear power plant using *RELAP*5/*Mod*3.2 computer code has been presented by Zare *et al.* [19]. The study aims to assess the accuracy of the *RELAP*5 model in predicting *NPP's* behavior during such accidents and evaluate the effectiveness of an automatic accident management algorithm in mitigating the consequences. Key findings include the study's simulation accuracy, the importance of an automatic accident management algorithm in mitigating the consequences of leak accidents, and the analysis's operational insights into *NPP's* dynamic response.

The existing studies provide valuable insights into *NPP* safety, including accident management and emergency operating procedures. However, there are still areas for further exploration. The existing studies emphasize the transition from operator-dependent interventions to passive safety systems in advanced *NPP* designs. Key areas for further investigation include refining simulation models, understanding the full extent of radiological consequences, improving emergency response strategies, developing comprehensive risk assessment methodologies, and conducting validation studies. *VVER*-1200 reactor requires a thorough analysis of accident management strategies for multifunctional transient scenarios to improve safety and minimize radioactive material release risks, while further research is needed to refine accident assessment models and understand *NPP's* dynamic response. The proposed study introduces a novel approach using *PCTRAN* simulation to replicate the complex dynamics of *VVER*-1200 under operational and multifunctional-transient conditions. It employs detailed modeling techniques to precisely simulate combined scenarios of *SGTR* and *AC* power loss transient scenarios, including reactor kinetics equations, thermal-hydraulic processes, and control systems. The study evaluates the performance of essential safety systems, such as emergency core cooling and high-pressure coolant injection, under proposed conditions. This assists to identify potential vulnerabilities and informs improvements in emergency response protocols. The study assesses the radiological consequences of *SGTR*, *AC* power loss and their combined scenarios, quantifying the release of radioactive materials that impact on public health, environment. This provides valuable contributions into the potential consequences of nuclear accidents and informs mitigation strategies. The study contributes significantly to the advancement of *NPP* safety by improving understanding of transient events, identifying safety vulnerabilities, and informing regulatory standards and guidelines. This enhances decision-making and emergency response planning and ensures that industry practices align with the latest scientific knowledge and best practices.

The manuscript is structured into five sections, as follows: section 1 provides a general introduction and background, encompassing the problem statement, objectives, and scope of the work as well as literature review of existing studies is discussed, aiming to identify research gaps and areas that form a solid foundation for the novelties and contributions of the proposed study. Section 2 describes the methodology and working procedures involved in simulating proposed transient events. Section 3 presents the results obtained and section 4 provides an analysis of the findings. Finally, section 5 offers conclusions and recommendations for future research work.

2. Methodology

In the methodology section, a detailed account of *VVER*-1200 reactor's general design, research methodology, data collection procedures, and overall organization is presented. It delves into the various conditions for *PCTRAN* simulation model, which are crucial for understanding reactor dynamics in different operational scenarios. We delve into the modeling and simulation techniques utilized in PCTran, emphasizing the safety measures applied to simulate *VVER*-1200 reactor dynamics, especially in relation to *SGTR* and *AC* power loss incidents. The methodology section offers a comprehensive overview of the approach to analyzing *VVER*-1200 reactor dynamics, emphasizing safety and operational stability in challenging scenarios.

2.1. *VVER*-1200 General Design

VVER-1200 plant is designed to optimize energy generation while prioritizing safety and reliability [20]. **Figure 1** represents the simplified layout of *VVER*-1200. It consists of a reactor core, primary circuit, steam generators, turbine hall, condenser, auxiliary systems, and control room. The reactor core houses nuclear fuel assemblies with control rod, while the primary circuit transfers heat from the reactor core to the steam generators. Steam generators transfer heat from the primary coolant to the secondary circuit, driving turbines to generate electricity. The turbine hall houses turbines and equipment for converting steam energy into mechanical energy. The condenser recycles steam back into water, while cooling towers cool and recycle water. Auxiliary systems support the operation of the main components, including emergency cooling, feedwater, and control systems. The control room houses operators and control systems responsible for monitoring and controlling plant parameters. Passive heat removal system (*PHRS*) is added to the steam generators [21], allowing natural steam circulation to be condensed by ambient air outside the containment. *VVER*-1200 reactor design increases the pressure of the primary circuit and steam generators [22], and the capacity of the main circulation pumps.

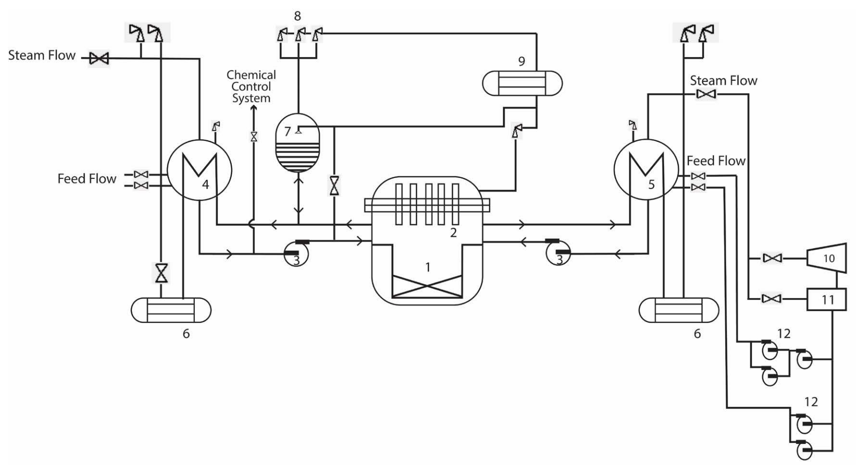


Figure 1. Simplified schematic diagram of *VVER*-1200 plant: 1-Reactor Core, 2-Control Rod, 3-Pump, 4-Steam Generator A, 5-Steam Generator B, 6-Passive heat removal system, 7-Pressurizer, 8-Pressurizer relief valve, 9-Bubble condenser, 10-Turbine, 11-Condenser, 12-Main Condensate pump.

2.2. Framework of the Study

The proposed study encompasses a series of sequential steps, commencing with the initiation of PCTRAN, *VVER*-1200 (*version* 1.0.0), followed by the initialization of

NPP operation at 100% power at the end of the cycle (*EOC*) as illustrated in **Figure 2**. *PCTRAN* simulation is designed to replicate the dynamics of *VVER*-1200 plant under various operational and transient conditions [23]. It can optimize emergency procedures, verify safety systems, assess regulatory compliance, and aid in research and development by identifying effective responses, design flaws, and enhance understanding of reactor dynamics [24] [25]. Subsequently, the simulation branches based on the operating conditions, leading to either normal operation or transient operations. In the case of normal operation, the process involves initial as well as boundary parameter setting (**Table 1**), selection of simulation time, and preparation of basic data. Conversely, transient operations encompass specific scenarios such as *AC* power loss, *SGTR*, and combination of both, each requiring distinct simulation modeling (**Table 2**) with respect to safety criteria (**Figure 3**) selection. The subsequent step involves the preparation of datasets, including basic data for normal operation and thermal-hydraulic as well as radiation data for transient operations. Final assessment is carried out, encompassing safety systems dynamics, reactor operation parameters, and radiation monitoring and consequences. Following this, the simulation results are compared, validated, and assessed for impact. This includes evaluating reactor coolant loop performance in transient conditions, and assessing transient event dynamics, considering temperature variances and reactivity alterations. *SGTR* combined with *AC* power loss in a nuclear power plant leads to severe consequences, including core cooling loss, reactor shutdown, and loss of emergency core cooling systems. *AC* power loss also impairs vital instrumentation and control systems, potentially causing fuel damage and radioactive release. Recovery and restoration can be challenging due to damage to electrical systems and safety equipment. Hence, simulation is crucial in understanding and mitigating *SGTR* and *AC* power loss scenarios in nuclear power plants. It provides a detailed analysis of plant systems, aids in emergency response planning, and helps evaluate mitigation strategies.

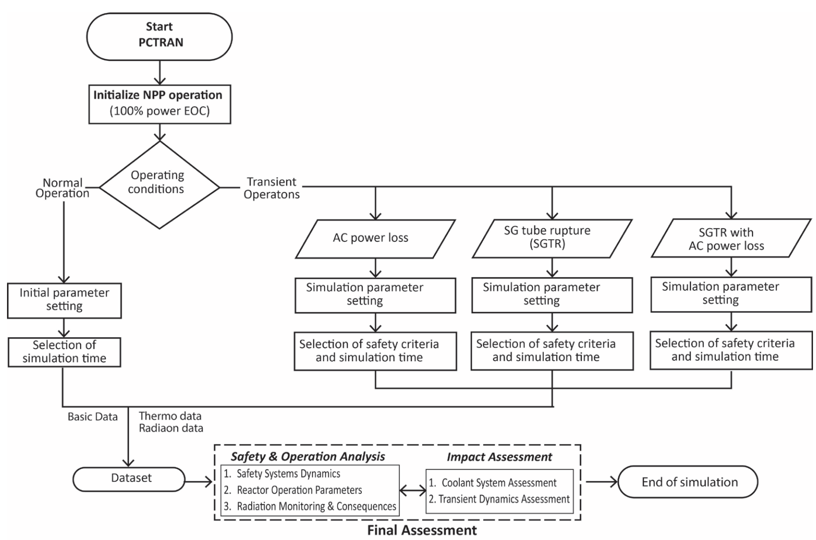


Figure 2. Framework of the study.

Table 1. Boundary conditions and initial conditions of *PCTRAN* simulations.

Parameter	Value	Dimension
Core Thermal Power	3200	MW
Initial *RCS* pressure	162	bar
Initial average *RCS* temperature	313.55	˚C
Total core flow rate	62,200	t/hr
Reactor trip pressure	180	bar
Total *RCS* heat input	16	MWt
Steam generator relief valve opening pressure	78	bar
Steam generator pressure at 100% power	74	bar
Total *RCS* volume excluding pressurizer	290	M^3
Steam generator water inventory	260,000	Kg
Steam generator relief valves total capacity	400	t/hr
Pressurizer volume	79	M^3

Table 2. Modelling of malfunctions and failures in *PCTRAN* for *VVER*-1200.

Case	Malfunction	Delay time	Failure fraction	Simulation time	Criteria	Transient Status
Case-1	*AC* power loss	10 sec	80%	1800 sec	-	Active
Case-2	*SG-A* tube rupture	10 sec	0.8	1800 sec	1 % of full tube rupture	Active
	SG-A tube rupture	10 sec	0.5	1800 sec	1 % of full tube rupture	Active
Case-3	*SGTR* with AC power loss	10 sec	0.8 fraction with 80% loss	1800 sec	1 % of full tube rupture	Active

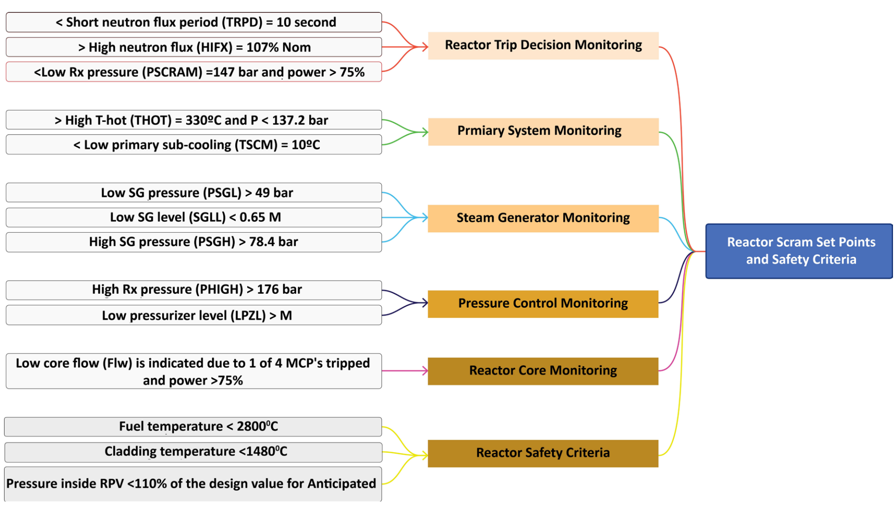


Figure 3. Safety criteria for simulation of *VVER*-1200.

2.3. Different Conditions for the PCTRAN Simulation Model

Simulating transient scenario such as *SGTR* and *AC* power loss scenario using

PCTRAN involves intricate modeling and analysis to ensure accurate representation of the nuclear power plant dynamics [26] [27]. At the core of these simulations are several fundamental equations and models that capture the behavior of various plant components and systems under extreme conditions. Firstly, the reactor kinetics equations play a pivotal role in understanding the transient response of the reactor core. Equation (1) depicts the changes in neutron population within the core concerning alterations in neutron flux and temperature. *PCTRAN* incorporates point kinetics models to simulate how the reactor core dynamically reacts during *AC* power loss and *SGTR* events.

$$\frac{dp}{dt} = \frac{\beta}{k_{eff}}\left(\bar{\phi} - \bar{\phi}_0\right) - \lambda\rho \quad (1)$$

where, ρ reactivity, β is effective delayed neutron fraction, k_{eff} is effective neutron multiplication factor, ϕ is relative neutron flux, λ is decay constant. Within the realm of heat transfer, *PCTRAN* employs equations to model thermal processes occurring within the reactor core, steam generators, and other plant components. Thermal-hydraulic models are utilized to characterize the behavior of the primary and secondary coolant systems. These models encompass mass balance equation (Equation (2)), energy balance equation (Equation (3)), and momentum balance equation (Equation (4)) to account for heat transfer, pressure fluctuations, and flow distribution across the reactor coolant loops and steam generator tubes. Parameters such as coolant temperature, pressure, flow rate, and heat transfer coefficients are crucial inputs in these equations.

$$\frac{dM}{dt} = \dot{m}_{in} - \dot{m}_{out} \quad (2)$$

$$\frac{dH}{dt} = \dot{Q}_{in} - \dot{Q}_{out} \quad (3)$$

$$\frac{dP}{dt} = \frac{\dot{m}_{in} v_{in} - \dot{m}_{out} v_{out}}{A} \quad (4)$$

Here, m_{in} is inflow mass, m_{out} is outflow mass rate, v_{in} is inflow velocity, v_{out} is outflow velocity, A is cross-sectional area, Q_{in} is inflow heat transfer rate, Q_{out} is outflow heat transfer rate. Central to simulating *SGTR* events are the incorporation of models representing the rupture or leakage of steam generator tubes with *AC* power loss. Equation (5) encompass the release of coolant inventory, loss of heat removal capability, and potential discharge of radioactive steam into the secondary containment. Factors such as tube rupture size, coolant flow rates, and secondary system response are carefully modeled.

$$\dot{m}_{leak} = C_d \sqrt{2\rho\left(P_{primary} - P_{secondary}\right)} \quad (5)$$

m_{leak} is mass flow of coolant leakage from ruptured steam generator tubes, C_d is discharge coefficient, ρ is Density of coolant, $P_{primary}$ is primary coolant pressure, $P_{secondary}$ is secondary (steam generator) pressure. Emergency core cooling system models are essential components of *PCTRAN* simulations, ensuring adequate cooling of the reactor core during transients. These models represent systems like core spray systems (Equation (6)) and, high-pressure injection (Equation (7)) simulating pump performance, flow distribution, and pressure-temperature correlations to maintain core cooling under adverse conditions.

$$\dot{m}_{core\_spray} = f_{core\_spray}\left(P_{core\_spray}, T_{core\_spray}\right) \tag{6}$$

$$\dot{m}_{HPI} = f_{HPI}\left(P_{HPI}, T_{HPI}\right) \tag{7}$$

Here, m_{core_spray} is mass flow of water injected by the c ore spray system, P_{core_spray} is pressure at the core spray injection point, T_{core_spray} is temperature at the core spray injection point. m_{HPI} is mass flow of water injected by the high-pressure injection system, P_{HPI} is pressure at the high-pressure injection point, T_{HPI} is temperature at the high-pressure injection point. Control system (Equation (7)-(8)), operator actions, and instrumentation and monitoring models complete the comprehensive suite of *PCTRAN* features. These elements collectively enable thorough simulations of *AC* power loss with *SGTR* scenarios, facilitating safety analyses, accident management strategies, and operator training in nuclear power plan.

$$\dot{\rho} = \frac{-\rho}{\tau} \tag{7}$$

$$\dot{P}_{depressurization} = -k_{dep} \times P_{depressurization} \tag{8}$$

Here, $P_{depressurization}$ is reactor pressure during emergency depressurization, k_{dep} is depressurization rate constant, τ is reactivity insertion time constant. **Table 1** presents a set of boundaries and initial conditions for *PCTRAN* simulations related to *VVER*-1200 reactor dynamics in scenarios involving *SGTR* and *AC* power loss. are based on the normal operating conditions of *VVER*-1200 reactor, ensuring that the simulations start from a steady-state that accurately reflects typical reactor behavior under standard operational parameters and is based on the normal operating conditions of *VVER*-1200 reactor [28]. Such parameters are typically sourced from safety analysis documents, such as the Preliminary Safety Analysis Report, which provide validated baseline conditions necessary for safety assessments. Deviations from these initial conditions trigger transient scenarios, and therefore, these values are crucial for accurately initiating the simulation process and analyzing reactor response to potential safety challenges.

Table 1 includes various parameters, their corresponding values, and dimensions. These parameters are crucial for conducting PCTRAN simulations, which are essential for understanding and predicting the behavior of the reactor under specific operational and safety scenarios. By incorporating these parameters into their analyses, scholars can gain insights into the thermal, hydraulic, and safety aspects of VVER-1200 reactors, contributing to the advancement of nuclear reactor safety and operational stability research.

2.4. Modelling Simulation in PCTran and Safety Criteria for Simulation of *VVER*-1200

The scenarios of system malfunction presented in **Table 2** come with individual, distinct case. Case-1 involves *AC* power loss at 10 sec, while Case-2 involves *SG-A* tube rupture. Case-3 involves a combined failure, with specific criteria. The equations and models mentioned above, along with appropriate safety criteria (**Figure 3**), boundary conditions and initial conditions (**Table 2**), form the basis for simulating *AC* power loss with *SGTR* scenario using *PCTRAN*. They enable the analysis of system behavior, identification of safety vulnerabilities, and evaluation of mitigation strategies to enhance the safety and reliability of *VVER*-1200.

Figure 3 outlines safety criteria in *PCTRAN* simulation, crucial parameters monitored in the reactor protection system panel. These set points indicate reactor safety and integrity, triggering emergency shutdown procedures. During simulation, if any of the monitored parameters breach their respective set points, it signifies abnormal conditions within reactor system. Abnormal conditions, such as short neutron flux periods, high neutron flux, low primary sub-cooling, and low steam generator pressure, require immediate action to ensure safety. Reactor scram set points in *PCTRAN* are crucial for ensuring safety and stability of nuclear reactors during scenarios such as *AC* power loss, steam generator tube rupture. These set points trigger automatic shutdown actions when certain parameters are exceeded, preventing unsafe operating conditions, and mitigating potential accidents. They are essential components of the reactor's safety systems, providing automated protection to prevent unsafe conditions and initiate necessary safety measures in abnormal operating conditions.

3. Results

In this section, we delve into the dynamics and control mechanisms of safety systems in *VVER*-1200 nuclear power plants. The research focuses on sequential simulated transient events, power generation dynamics, and control during safety system responses. Key parameters influencing *VVER*-1200 operation, temperature measurements in the reactor coolant system, radiation monitoring assessments, and isotope decay characteristics under transient conditions are thoroughly examined. Furthermore, the study includes a validation and impact analysis section that evaluates the performance of the reactor coolant loop. This assessment involves comparing simulated results with safety protocols to ensure compliance. Additionally, an evaluation of transient conditions within thermal hydraulic systems and safety responses is conducted to enhance understanding and optimize safety measures in *VVER*-1200 nuclear power plants.

3.1. Simulated Results

The scenario outlined in **Table 3** depicts sequence of transient events within *VVER*-1200, highlighting occurrences related to reactor operation, turbine functionality, and associated systems during a specified time frame. In the initial phase, at time 0.5 *sec*, malfunction with a fractional impact of 80.0% occurs, influencing both *AC* power loss and *SGTR*. This malfunction, characterized by its severity, initiates a consequential chain of events within the plant. A significant event unfolds at time 12 *sec*, marked by the trip of reactor coolant pumps (*RCPs*) A and B, leading to an immediate Reactor Scram. This event disrupts normal operations, impacting both *AC* power loss and *SGTR* systems, indicating the criticality of the situation. Shortly after, at time 12.5 *sec*, a change in turbine control valve (*TCV*) position triggers turbine trip, further complicating the operational environment. This event, like its predecessors, influences both *AC* power loss and *SGTR*. Between times 20 and 21 *sec*, adjustments in the positions of safety relief valves (*SRVs*) for various systems take place, affecting *AC* power loss and *SGTR* differently. These changes represent efforts to mitigate potential hazards and maintain system integrity. A critical event occurs at time 29.5 *sec*, as a reactor scram is initiated due to high *SG* pressure and low secondary coolant system

Table 3. Sequential order of the simulated transient events.

Time	Transient events		
	***AC* power loss**	***SGTR***	***SGTR* with *AC* power loss**
0	100% End of cycle (*EOC*)	100% *EOC*	100% *EOC*
0.5	Malfunction (*AC* Power Loss) Fraction = 80.0 %	Malfunction (*SGTR*) Fraction = 00.8 %	Malfunction (*AC* Power Loss) Fraction = 80.0% Malfunction (*SGTR*) Fraction = 00.8%
12	RCP-A & B trip	Scram Low *RC* Flow, Reactor Scram	Scram Low *RC* Flow, MFW Pumps trip
13	Malfunction (Load Rejection) Fraction = 60.0 %	Malfunction (Load Rejection) Fraction = 60.0 %	Malfunction (Load Rejection) Fraction = 60.0%
12.5	*TCV* valve 1 position change, Turbine trip	*TBV* valve 1 position change	*TCV* valve 1 position change, Turbine trip
20	*SG SRV* 1 position change	*PZR* spray valve 1 position change	*PZR* safety relief valve 1 position change
21	-	*PZR* Backup Heater Capacity Change	*PZR* Safety Relief Valve 2 Position Change
21 - 28	*SG SRV* 2 - 3 position change	*SG SRV* 1 - 2 position change	*SG SRV* 2 - 4 position change
29.5	Scram High *SG* Press 88.00 bar	*PZR* proportional heater capacity change	*PZR* safety relief valve 2 position change scram low SCM 10.0 C
32.5	Reactor scram	-	Reactor Scram
54	Low *SG* level 2.1 m, D/G A starts	Low *SG* level 2.1 m, D/G A starts, FW isolation on low 281C	Low *SG* level 2.1 m, D/G A starts 60.0 *sec* delay
71.5	*TDAFW* pump 1 - 2 position change	*PZR* spray valve 1 position change	*TDAFW* pump 1 - 2 position change
132 - 1673	*SGSR* 1 - 2 position change	*SGSR* 1 - 4 position change	*SGSR* 1 - 2 position change

(*SCM*) temperature. This event underscores the importance of safety protocols and their impact on *AC* power loss and *SGTR*. At time 54 *sec*, a Low *SG* Level event prompts the activation of diesel generators (*DGs*), further influencing *AC* power loss and *SGTR*. This response illustrates the plant's reliance on backup systems during emergencies. Another notable occurrence at time 71.5 *sec* involves a change in the position of turbine driven auxiliary feedwater (*TDAFW*) pumps, affecting *AC* power loss and *SGTR*. Such adjustments are vital for maintaining essential functions within the plant. Throughout the duration of the scenario, between times 132 and 1673 *sec*, there are multiple changes in the positions of safety relief valves (*SRVs*) for steam generators. These adjustments, occurring over an extended period, reflect ongoing efforts to manage operational conditions and safeguard plant assets.

ECCS is crucial for nuclear power plants' safety and integrity, especially during coolant loss accidents. Initially, *ECCS* maintains core cooling consistently, preventing overheating and potential meltdown. However, it fails to operate after a certain duration without electrical power, highlighting a vulnerability (**Figure 4(a)**). *ECCS* exhibits reliability during steam generator tube rupture events, ensuring stable core cooling. However, the combined occurrence of *AC* power loss and *SGTR* shows a gradual degradation in *ECCS* performance over time, indicating potential challenges in prolonged emergency situations. This highlights the need for continuous monitoring, maintenance, and testing protocols to identify and address vulnerabilities promptly. *HPCI* is a safety system that provides emergency core cooling by injecting high-pressure coolant directly into the reactor vessel [29]. It is activated following *SGTR* events, similar to *ECCS* (**Figure 4(b)**). *HPCI* effectively maintains reactor coolant pressure and temperature within acceptable limits, contributing to reactor safety during *SGTR* events. It serves as a redundant safety measure alongside *ECCS* [30], providing additional protection against core overheating and maintaining reactor stability in case of coolant loss accidents. **Figure 4(c)** provides the behavior of reactor core's dynamic, in maintaining safe and stable operation through effective reactivity control and emergency response measures. Reactivity rod (% dk/k) is crucial for understanding safety and stability of *VVER*-1200 under various conditions [31]. At the beginning, all scenarios show a stable reactivity level of 0%, indicating the reactor is operating within desired parameters. However, as the simulation progresses, reactivity levels vary, especially in response to *AC* power loss, *SGTR*, and their combination. *AC* power loss maintains a constant reactivity level, while *SGTR* causes a gradual decrease, indicating a rise in reactivity. Simultaneous *AC* power loss and *SGTR* pose a heightened risk to reactor safety, requiring immediate activation of emergency protocols and backup systems. Pressurizer level (m) in reactor system is analyzed for safety and thermal hydraulics. It remains stable around at 8.17 m in *AC* power loss scenario, indicating effective control and mitigation of disturbances (**Figure 4(d)**). However, in *SGTR* scenario, pressurizer level decreases slightly, suggesting compensatory action. The combined effect of tube rupture and

AC power loss requires more compensatory action. A stable pressurizer level indicates effective control and mitigation, while significant deviations may indicate challenges in maintaining reactor stability and safety [32].

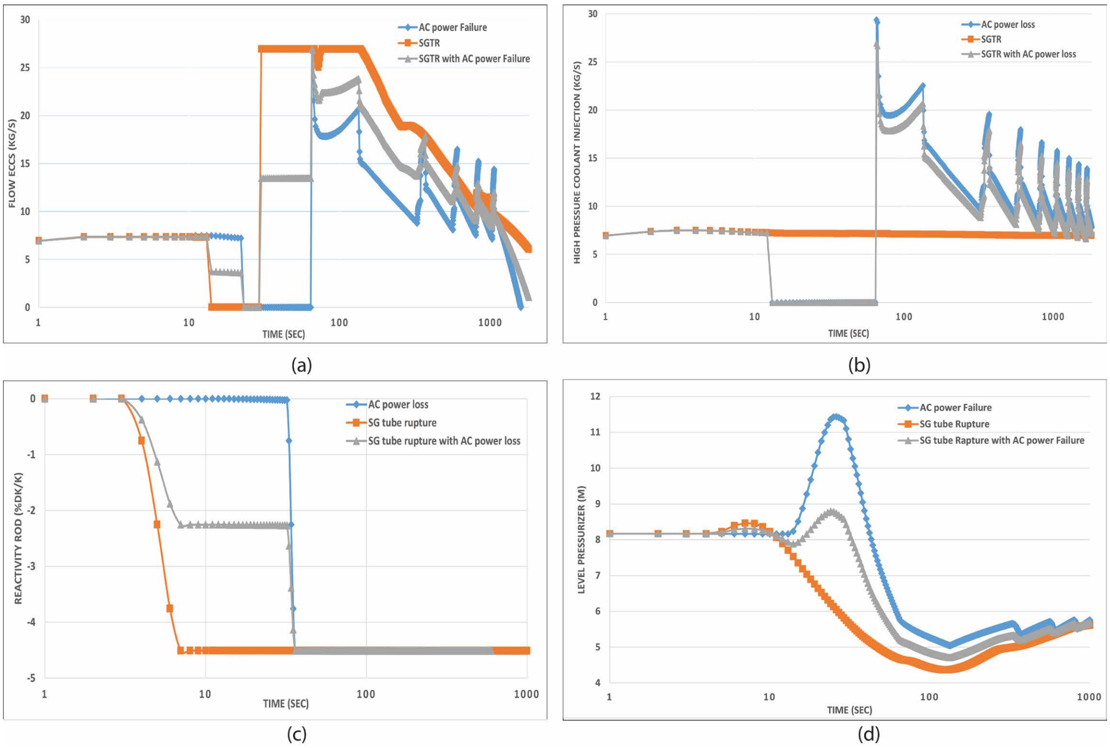


Figure 4. Dynamics and control of safety systems: (a) Flow of ECCS (kg/sec) (b) High pressure coolant injection (kg/sec) (c) Reactivity Rod (% dk/k) (d) pressurizer level (m).

In this study, *VVER*-1200 maintains a consistent thermal power output of 3200 MW under standard conditions (**Figure 5(a)**) during the proposed scenario. However, it fluctuates slightly when *AC* power loss occurs, indicating adaptive mechanisms. The most notable deviation is in the event of *SGTR*, where thermal power declines over time, indicating safety protocols. The reactor's response is more pronounced when facing both events. **Figure 5(b)** reveals dynamics of reactor's nuclear power flux under different operational scenarios. During normal operation, nuclear flux remains steady at 100% of nominal value. However, when *AC* power loss occurs, it fluctuates slightly, demonstrating its adaptability. The reactor's safety protocols adjust to the anomaly, while combined effect of *SGTR* and *AC* power loss results in a more pronounced decrease. **Figure 5(c)** provides detailed interpretation of reactivity fuel behavior, focusing on the reactivity coefficient expressed as a percentage (% dk/k). It reveals distinct trends and responses to conditions such as *AC* power loss, *SGTR*, and the combination of both. Under normal operation, reactivity fuel remains stable at 0%. However, abnormal situations can cause deviations. During *AC* power loss, reactivity fuel percentage increases gradually, possibly due to altered cooling mechanisms or neutron moderation changes. SGTR results in a rapid increase, likely due to disturbances in coolant flow or core geometry. Combined *SGTR* and *AC* power loss exacerbates the reactivity response, underscoring the risk and urgency of simultaneous failures. **Figure 5(d)** presents the heat removal process of *SG* in *VVER*-1200, revealing a stable thermal equilibrium under normal conditions. However, when *AC* power loss occurs, heat removal fluctuates, indicating a diminishing ability to dissipate heat without electrical power. *SGTR* results in a rapid and significant reduction in heat removal, indicating a disruption in the cooling system's functionality. The combined effect of *AC* power loss and *SGTR* increases operational challenges and safety risks.

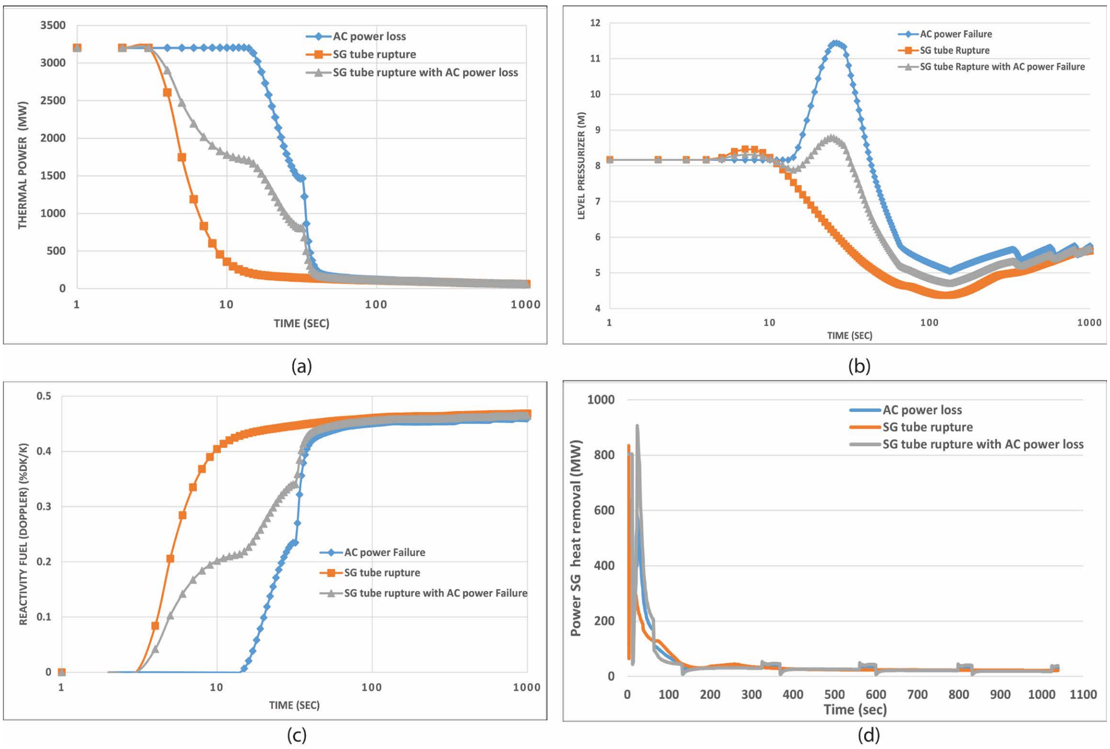


Figure 5. Power generation dynamics and control in safety system response: (a) Thermal power output in MW(b) Nuclear power flux (%) (c) Fuel Reactivity Rod (% dk/k) (d) Power SG heat removal in MW.

Maintaining *RCS* pressure within design limits is crucial for ensuring integrity of reactor system and preventing accidents such as fuel damage or core meltdown [33]. Deviations from normal pressure conditions, especially significant increases resulting from tube ruptures, signal potential safety risks that require immediate attention and intervention. In case of *AC* power loss, *RCS* pressure remains relatively stable over time, indicating that system can manage heat generated within safe limits even without external power supply (**Figure 6(a)**). When *SGTR* occurs, there is a noticeable increase in *RCS* pressure, which is indicative of a loss of coolant. This scenario requires careful monitoring and potential activation of safety systems to mitigate the consequences. The combination of *SGTR* with *AC* power loss results in a more complex situation, with initial rapid increase in *RCS* pressure similar to *SGTR* scenario. As time progresses, pressure tends to stabilize at a higher level, suggesting that the combined scenario exacerbates pressure buildup.

Specific enthalpy is a crucial parameter in thermodynamics and thermal hydraulics, indicating the total energy content of a fluid per unit mass. The specific enthalpy of pressurizer changes is shown in **Figure 6(b)**. In *AC* power loss scenarios, pressurizer remains relatively stable over time, suggesting it is equipped with sufficient passive heat removal mechanisms or backup power sources. When rupture occurs in *SG* tube, specific enthalpy fluctuates, initially increasing due to the release of heat from the primary loop. However, over time, it stabilizes and decreases as the RCS compensates for the loss. In *SGTR* with *AC* power loss scenarios, the specific enthalpy stabilizes over time, suggesting the reactor can maintain its safety functions and thermal equilibrium, albeit with some transient effects. Departure from Nucleate Boiling Ratio (*DNBR*) is a safety parameter used in nuclear reactor analysis, [34], indicating the margin to the onset of nucleate boiling in the fuel assemblies [35]. If *DNBR* drops below a certain threshold, it indicates a potential for boiling crisis, which could lead to fuel damage. **Figure 6(c)** signifies the changes of *DNBR* during three scenarios. *DNBR* remain relatively stable over time, suggesting a stable reactor core from a thermal hydraulics perspective. However, there is a slight decreasing trend in *DNBR* over time, indicating a gradual decrease in safety margin. In *SGTR* scenario, *DNBR* values exhibit a significant and rapid decrease initially, indicating a sudden reduction in safety margin. As time progresses, *DNBR* continue to decrease, albeit at a slower rate, indicating a gradual deterioration in thermal conditions. *SGTR* can occur in different fractions (80% and 50%), with higher fractions causing more significant changes in flow rate according to **Figure 6(d)**. Initial drop-in flow rate and gradual recovery may be due to compensatory mechanisms or operator adjustments. This can lead to a loss of *SG* level, a safety concern in nuclear power plants. The combination of *AC* power loss and *SGTR* exacerbates the challenges, with a more abrupt initial drop and slower recovery.

The behavior of *RCS* under various operational and accident scenarios is illustrated in **Figure 7(a)**, contributing to the validation of thermal-hydraulic models and simulations. *RCS* temperature changes under different scenarios reveals that it remains relatively stable around 313.5˚C initially, indicating that natural circulation and passive cooling systems effectively remove heat from the reactor core. However, as time progresses, the temperature decreases, indicating gradual decay heat removal. When *SG* tube rupture occurs, *RCS* temperature increases steadily over time, indicating a potential loss of coolant. The rapid temperature increase during *SGTR* with *AC* power loss highlights the severe impact of simultaneous failures on reactor safety and the urgency of implementing emergency cooling measures. Reactor safety and thermal hydraulic behavior under transient scenarios are perceived by changes in pressurizer temperature trends. **Figure 7(b)** shows that pressurizer temperature decreases gradually over time in the event of *AC* power loss, as there is no external source of heat to maintain the temperature. The temperature increases significantly in the event of *SGTR* due to the release of hot primary coolant. The temperature decreases gradually as the system loses heat due

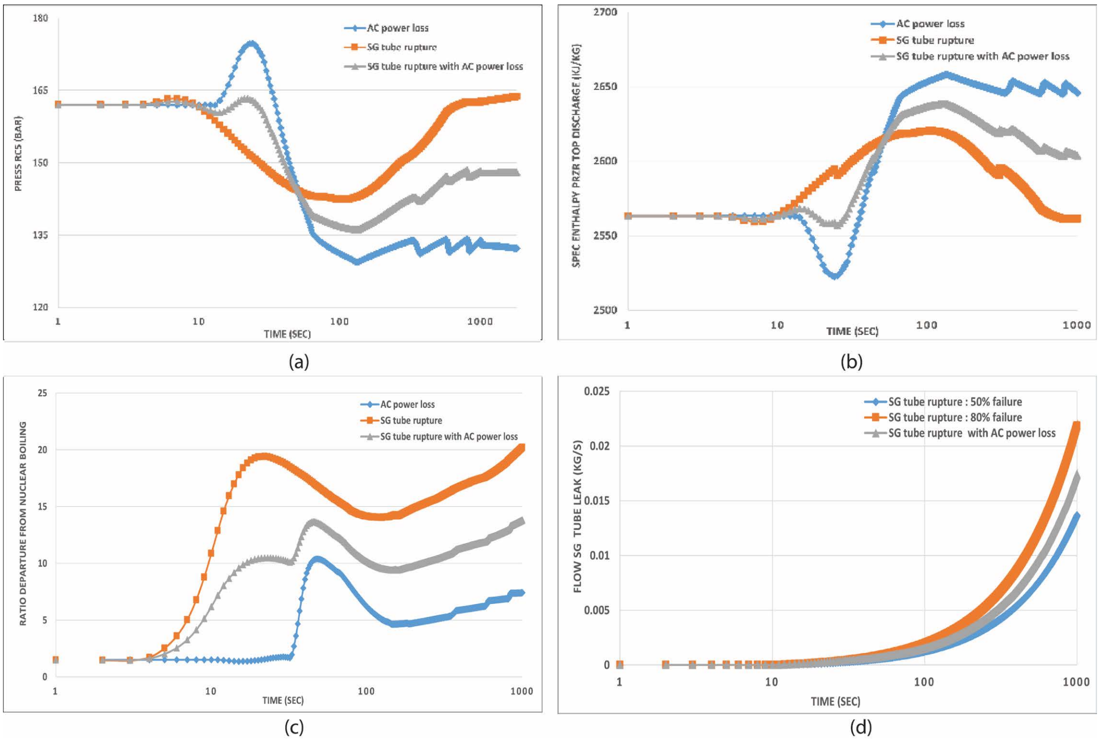


Figure 6. Key Parameters in *VVER*-1200 operation: (a) *RCS* pressure in bar (b) Specific enthalpy of pressurizer in KJ/KG (c) Fuel reactivity rod (% dk/k) (d) *DNBR*.

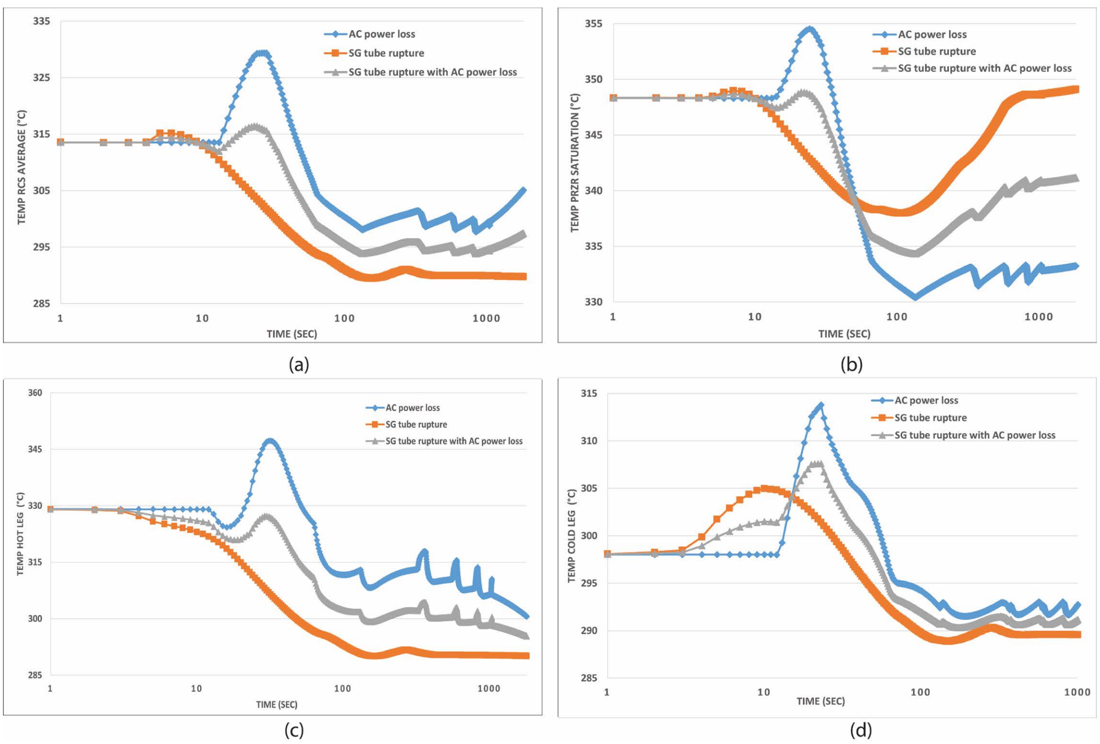


Figure 7. Temperature measurements in reactor coolant system in ˚C: (a) *RCS* temperature (b) Pressurizer temperature (c) Temperature in hot leg (d) Temperature in cold leg.

to *AC* power loss. Monitoring these trends help to assess the severity of the situation and take appropriate actions to mitigate risks. *SGTR* is a more immediate safety concern than *AC* power loss alone, as it leads to a rapid increase in pressurizer temperature. Understanding these changes is crucial for designing and operating nuclear power plants safely and efficiently. Thus, the analysis of pressurizer temperature aids in developing robust safety protocols and emergency response strategies. Figure 8(c) shows significant rise in temperature in hot leg, indicating potential breach in the primary coolant system and potential overheating. Figure 8(d) in cold leg shows a different trend, suggesting that the effect of *SGTR* is exacerbated when *AC* power loss is considered. Comparative analysis of two Figure 7(c) and Figure 7(d) provide the differences in temperature behavior and system response under different conditions, highlighting the importance of maintaining steam generator tube integrity and emergency cooling systems.

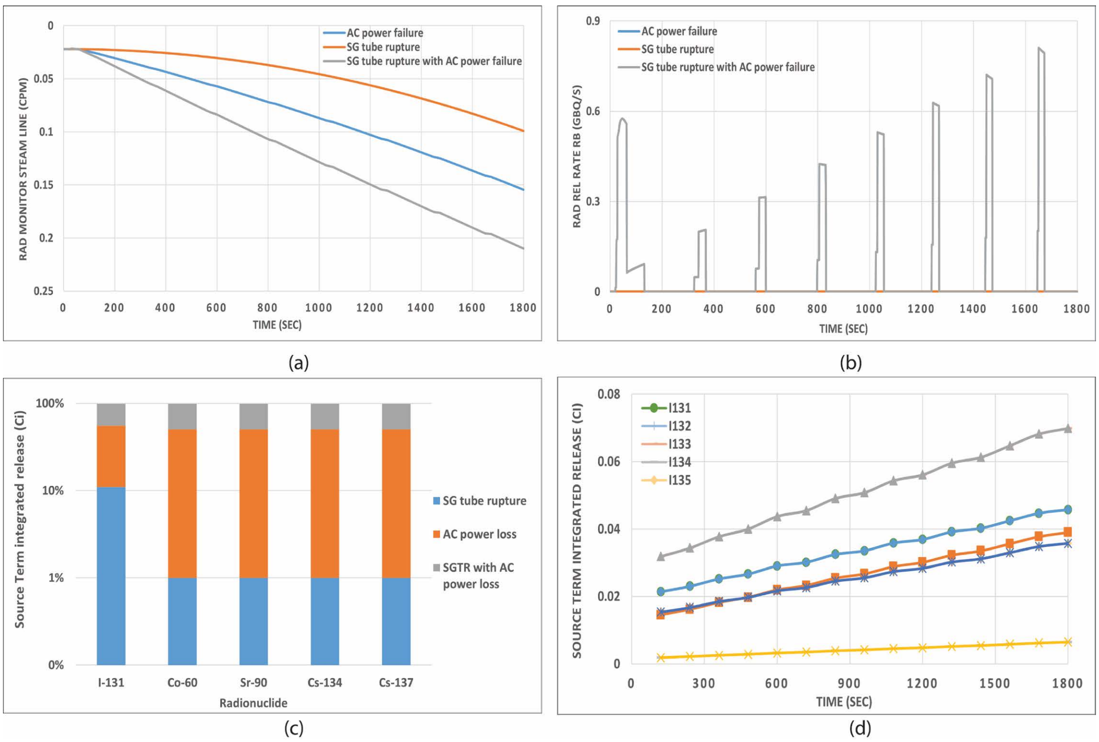


Figure 8. Radiation monitoring assessment: (a) Radiation levels in steam line (b) Measurements rates of radiation over time in *GBq*/*s* (c) Consequential event in terms of source term integrated releases (d) Cumulative release of isotope.

Figure 8(a) displays the radiation levels in the steam line measured in counts per minute (*CPM*) with time. The scenarios have minimal radiation release, while *SGTR* results in a gradual increase in radiation levels. *SGTR* with *AC* power loss combines the effects of both events, with higher levels compared to *SGTR* alone. The severity and rate of increase vary depending on the specific circumstances. Rupture in *SG* tube, especially when combined with an *AC* power loss, can lead to significant radiation release, posing potential health and safety risks. Figure 8(b) represents measurements rates related to radiation over time under different conditions in *GBq/s*.

The presence of *AC* power loss exacerbates the release of radioactive material, causing contamination of the surrounding environment and health risks to personnel and the public. The combination of *SGTR* and *AC* power loss results in an increased release of radioactive material, necessitating intensified emergency response efforts. The most consequential event is the scenario of *SGTR* with *AC* power loss, which is likely to have the most severe consequences due to its higher release magnitudes, especially *I*-131, which poses significant risks to public health [36].

According to **Figure 8(c)**, *I*-131 is the predominant radioisotope released during transient conditions. *I*-131 is typically associated with thyroid cancer due to its high uptake in the thyroid gland [37]. Other isotopes, such as *Co*-60, *Sr*-90, *Cs*-134 and *Cs*-137 also pose significant health risks due to their radioactive properties [38], [39]. The dominance of *I*-131 in all scenarios highlights the importance of monitoring and mitigating its release during nuclear incidents to minimize health impacts, particularly thyroid cancer incidence. **Figure 8(d)** reveals the cumulative release of Iodine isotopes (*I*-131, *I*-132, *I*-133, *I*-134, *I*-135) under different scenarios. **Figure 8(d)** represents the cumulative release of each isotope over time. As the time progresses, more of each isotope is released into the environment due to ongoing processes like radioactive decay and potential spreading of contaminants. The release patterns of different isotopes have been compared with respect to decay mode such as alpha (α) decay, beta decay (β) and half-life to understand relative impact of each scenario on the release of specific isotopes. **Table 4** exhibit diverse characteristics that contribute to their potential impact in different scenarios.

Table 4. Assessment of isotope decay characteristics in transient conditions.

Isotopes	Half-life	Decay mode	Average source term activity (*Ci*)		
			AC power loss	*SGTR*	*SGTR* with *AC* power loss
Strontium-90	28.79 years	β decay to Yttrium-90	5.28×10^{-8}	1.06×10^{-9}	5.27×10^{-8}
Cesium-137	30.17 years	β and gamma radiation	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Iodine-131	8.02 days	β decay to Xenon-131	0.033767	0.010643	0.033766102
Plutonium-239	24,110 years	α decay to Uranium-235	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Plutonium-240	6560 years	α decay to Uranium-236	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Plutonium-238	87.7 years	α decay to Uranium-234	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Americium-241	432.2 years	α decay to Neptunium-237	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Curium-244	18.1 years	α decay to Plutonium-240	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}

Continued

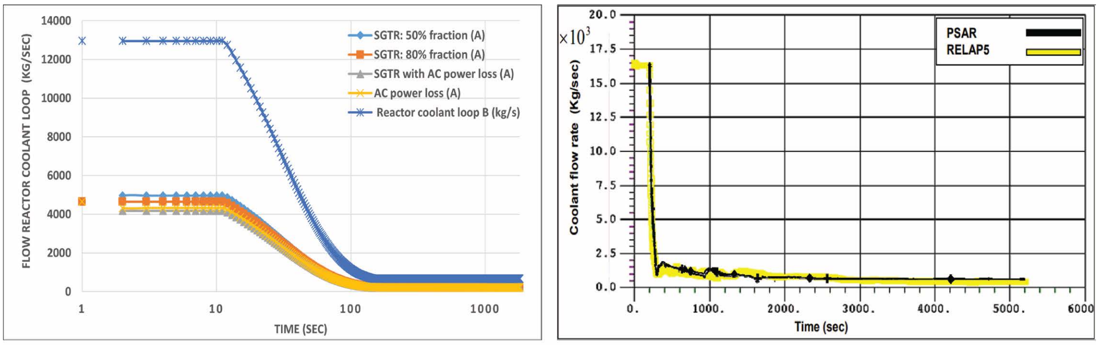
Tellurium-132	3.2 days	β decay to Iodine-132	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}
Ruthenium-106	1.01 years	β decay to Rhodium-106	5.28×10^{-8}	1.06×10^{-9}	5.28×10^{-8}

Among these isotopes, *I*-131 stands out as the most severe accident marker with an average source term activity of 3.3766%. This high activity level is attributed to its relatively short half-life of 8.02 days, which results in rapid decay and emission of beta radiation during nuclear accidents. In scenarios like *AC* power loss, the release of *I*-131 exacerbate the risks due to the absence of power for safety systems. In *SGTR* scenarios, the release compromises containment integrity and allow radioactive contamination. Furthermore, the decay mode of *I*-131, leading to *Xn*-131, contributes to its significant radioactive release and potential health risks. In contrast, isotopes with longer half-lives, such as *Sr*-90and *Cs*-137, although emitting beta and gamma radiation, have lower average source term activities, indicating comparatively lower severity in terms of their consequences.

3.2. Validation and Impact Analysis

Reactor coolant loop flow reveals various scenarios of *SGTR* (**Figure 9(a)**) events, such as 50% fraction, 80% fraction, and *SGTR* with *AC* power loss. These events pose significant safety concerns in nuclear reactors, as they lead to the release of radioactive materials and potential loss of coolant accidents. *AC* power loss also affects reactor safety, as it leads to the failure of essential safety systems. The reactor coolant loop flow rate is crucial for maintaining proper thermal hydraulics and cooling within the reactor core [40]. **Figure 9(b)** show variations in flow rates corresponding to different *SGTR* scenarios, with the specific impact depending on the severity of the rupture and the effectiveness of emergency systems. *AC* power loss scenarios also affect the coolant flow rate, with the potential failure of pumps and other systems dependent on electrical power. The findings from the flow reactor coolant loop are consistent with the assumptions outlined in the preliminary safety analysis report (*PSAR*) and another simulation performed by *RELAP*-5 [19] as outlined in **Figure 9(b)**. The reactor coolant loop maintains steady-state operation under normal conditions, indicating its ability to maintain these conditions. It responds appropriately to transient events like *SGTR* and *AC* power loss, demonstrating appropriate responses. The reactor coolant loop maintains adequate thermal hydraulics performance, preventing overheating and maintaining safe operating temperatures. Despite fluctuations in flow rates due to transient events, safety systems, such as emergency cooling mechanisms, are effectively activated and contribute to mitigating transient events.

Valuable perception into transient conditions, methodologies, and impact comparisons is provided by the comparison between existing and proposed studies within the thermal-hydraulic and safety systems aspects. **Table 5** offers a detailed comparison of thermal-hydraulic and safety system responses (*RCS* pressure, average *RCS* coolant temperature, pressurizer variations etc.) during transient conditions, highlighting critical aspects of reactor behavior. The proposed study stands out for its comprehensive evaluation of the system's response under realistic and complex conditions, addressing critical scenarios more holistically than previous studies. Utilizing *PCTRAN* simulation on *VVER*-1200 reactor, the study demonstrates improved responses in thermal-hydraulic and safety



(a) (b)

Figure 9. Assessment of reactor coolant loop performance (a) Simulated results (b) Performance in accordance with safety protocol.

Table 5. Evaluation of transient conditions within thermal hydraulic and safety system responses.

Study	Transient condition	Method	Impact comparison		
			RCS Pressure variation, bar	Average *RCS* coolant temperature, ˚C	Pressurizer level Variation, m
Matejovic *et al.* [15]	Station blackout	RELAP5/ Mod 3.2	115 - 122.5	260 - 340	11 - 13.5
Hossen *et al.* [16]	Steam Line Break	*PCTRAN* Demo version 1.2.0	160 - 170	240 - 323.38	11.20 - 13.43
Kim *et al.* [41]	*SGTR* Accident with Long-Term SBO	MELCOR code	110 - 170	-	-
Esfandiari *et al.* [17]	Loss of offsite power	RELAP5 code	140 - 160	291.0 (core inlet) 321.0 (core outlet)	6.5 - 13
Zare et al [19]	*SGTR*	RELAP5, *PSAR*	80 - 160	570 - 480 (core inlet) 600 - 500 (core outlet)	7.80
	SGTR (80% tube rupture)		135 - 165	289.47 - 315.16	5 - 8.3
This study	*AC* power loss	PCTRAN, *VVER*-1200 (version 1.0.0)	140 - 162	297.74 - 329.35	11.5 - 5
	SGTR with *AC* power loss		135 - 165	297.74 - 329.35	6 - 8.8

system parameters (such as pressure, coolant temperature, pressurizer level variations etc.) during transient scenarios. In the sphere of impact assessment, the proposed study provides notable advantages, particularly when considering the thermal-hydraulic and safety system aspects. Primarily, the study encompasses a combined transient condition involving *SGTR* and *AC* power loss. This scenario represents a critical and multifaceted challenge for safety systems, necessitating a comprehensive approach to evaluate the system's response under realistic and complex conditions. Moreover, the utilization of *PCTRAN* simulation as primary methodology for the proposed study offers a contemporary and sophisticated simulation platform which is capable of providing intricate and precise insights into system behavior. The ability to simulate complex scenarios with high fidelity is of paramount importance for comprehending the intricate interactions between thermal-hydraulic dynamics and safety system responses.

Notably, the study records lower average coolant temperatures and pressurizer level variations compared to its counterparts, indicating its effectiveness in assessing and managing safety risks within the thermal-hydraulic and safety system framework. Additionally, the outcomes are in line with the safety criteria and reactor scram setpoint list as indicated in **Figure 3** that show the efficacy of the proposed study in mitigating potential safety risks and upholding system stability under challenging conditions. However, deviations exist within the proposed study, which can be attributed to various factors including reactor design disparities, differences in simulation methodologies, boundary conditions, and modeling assumptions. The choice of simulation code and techniques holds significant sway over the results, potentially leading to discrepancies in accuracy and resolution. Furthermore, variations in boundary conditions and initial assumptions play a role in shaping the transient behavior of the reactor. Additionally, modeling assumptions and parameters introduce uncertainties, contributing to divergent outcomes. The reliability of results is influenced by the extent of validation and verification performed. Finally, the specific transient scenarios considered in each study can also impact the results, further emphasizing the complexity of transient analysis in nuclear power plants.

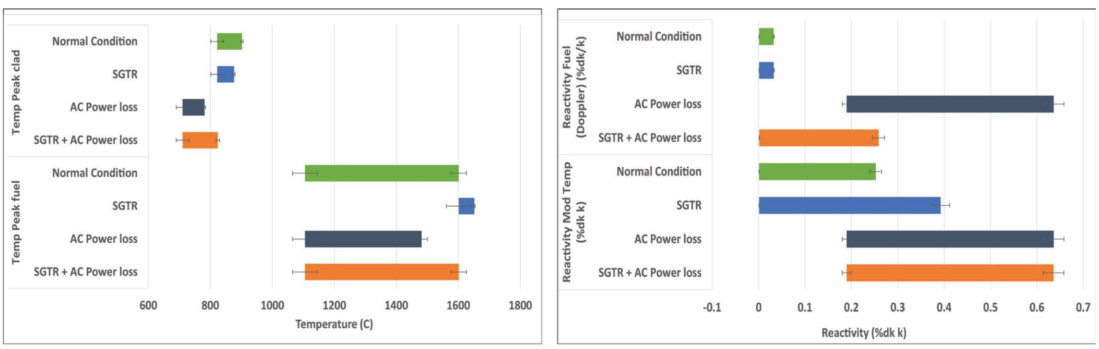
4. Discussion

The utilization of *PCTRAN* simulation in the exploration of *VVER*-1200 plant dynamics under multifunctional transient conditions aimed to gain a comprehensive understanding of reactor kinetics, heat transfer, and emergency core cooling systems. This simulation framework yielded valuable insights into reactor behavior, safety system efficacy, and radiological consequences. This study emphasized the adaptive mechanisms and safety protocols inherent in nuclear reactor systems, particularly noting the heightened reactor response when confronted with simultaneous events. Furthermore, the study conducted a comparative analysis of existing and proposed thermal-hydraulic and safety system perspectives during transient conditions, shedding light on critical aspects of reactor behavior.

Notably, the study observed lower average coolant temperatures and pressurizer level variations, indicating the effectiveness of the simulation in evaluating and managing safety risks within the thermal-hydraulic and safety system framework. Key discussion can be summarizing as follows:

- *ECCS* initially ensures stable core cooling but exhibits vulnerability to prolonged electrical power loss, emphasizing the need for continuous monitoring and maintenance. Similarly, *HPCI* effectively maintains reactor coolant pressure and temperature during *SGTR* events, serving as a redundant safety measure alongside *ECCS*. Behavior of reactivity rod reveals dynamic reactivity levels under different conditions, with *AC* power loss causing a constant reactivity level, *SGTR* leading to a gradual increase, and their combination posing heightened risks.
- Dynamic behavior under multifunctional transient scenarios highlights its adaptive mechanisms and safety protocols. The reactor maintains consistent thermal power output under standard conditions but exhibits slight fluctuations during *AC* power loss. It also shows effective safety measures in the event of *SGTR*. The reactor's response is more pronounced when facing both events simultaneously. Its sensitivity to abnormal conditions is evident in deviations during *AC* power loss and *SGTR*. Reliable cooling mechanisms are crucial for reactor safety and efficiency.
- The study emphasizes the importance of understanding *RCS* behavior for reactor safety analysis. Maintaining *RCS* pressure within design limits is crucial to prevent accidents like fuel damage or core meltdown. Deviations from normal pressure conditions, especially tube ruptures, indicate potential safety risks. The system can manage heat safely without external power, but careful monitoring is necessary for stability and safety.
- The study emphasizes significant thermal behavior and safety considerations of nuclear reactor systems under various scenarios. The submerged fuel's temperature remains stable at 800˚C, indicating a shutdown scenario. The *SG* level remains stable at 2.40 m, indicating safe water level even with *AC* power loss. The reactor coolant loop maintains steady-state operation under normal conditions and responds appropriately to transient events like *SGTR* and *AC* power loss.
- The dynamics of radiation release and associated health risks in nuclear reactor accidents are evident. Under normal conditions, there is minimal release, but during *SGTR* events, the release increases. When combined with *AC* power loss, the radiation levels are higher, particularly with *I*-131, which is linked to thyroid cancer. Monitoring and mitigating *I*-131 release is crucial to minimize health impacts in such scenarios.

The study examines the deviation of parameters, with a specific focus on the peak temperature of fuel and clad, from normal operation to gain insight into the transient severity on *VVER*-1200 (**Figure 10(a)**). Additionally, it analyzes the reactivity restraint temperature and reactive fuel (**Figure 10(b)**) to provide a comprehensive understanding of the reactor's behavior under varying conditions. The severity of the transient conditions is evaluated by comparison with normal operating conditions. *VVER*-1200 reactor's transient dynamics are significantly influenced by parameters such as fuel temperature, clad temperature, reactivity moderation temperature, and reactivity fuel (doppler effect). Fuel temperature is crucial for preventing fuel swelling, pellet-cladding interaction, and even fuel failure. Clad temperature is vital for maintaining the integrity of fuel rods and can spike rapidly during transients like loss of coolant accidents [42]. Reactivity moderation temperature affects the neutron flux distribution and reactivity of the core, potentially leading to power excursions or instability [43]. The Doppler effect, which changes the resonance absorption cross-sections of certain isotopes, is essential for accurately predicting reactor behavior during transient events and ensuring stable operation. Temperature peak fuel ranges from 304.72˚C to 800.54˚C, while temperature peak clad ranges from 410.83˚C to 411.02˚C. *SGTR* event results in a slight increase in temperature beyond these ranges, posing risks such as fuel overheating and potential damage to the reactor core. Conversely, during *AC* power loss, the temperature drops to 298.48˚C, significantly lower than the normal operating range. This lower temperature impacts the integrity of the clad and increase the risk of embrittlement or other structural issues. Reactivity moderation temperature remains within the range of −0.256% dk/k to 0.189% dk/k, indicating a slight increase in reactivity during *SGTR* event compared to normal conditions. However, these changes are relatively small and manageable within the safety margins of the reactor. The reactivity fuel range remains consistent across conditions, with a notable decrease in reactivity during *AC* power loss, potentially necessitating compensatory measures to maintain reactor stability.



(a) (b)

Figure 10. Assessing transient event dynamics in *VVER*-1200: (a) Temperature variances; (b) Reactivity alterations.

SGTR event demonstrates moderate deviations compared to normal conditions, leading to significantly lower severity transient condition compared to the other conditions. *AC* power loss also results in elevated temperatures and reactivity changes, albeit to a lesser extent compared to *SGTR*. This indicated that *AC* power loss also leads to significant deviations, particularly in reactivity-related parameters, from the normal condition. The combination of *SGTR* and *AC* power loss presents significant deviations from normal operating conditions, particularly in temperature ranges for fuel and clad, indicating the severity of this transient condition compared to normal conditions. Based on these deviations, it's evident that combination of *SGTR* and *AC* power loss presents a high degree of severity in terms of temperature and reactivity. During transient conditions, safety systems such as *ECCS*, pressurizer system, and *HPCI* collaborate to maintain reactor safety. *ECCS* cools down the reactor core, adjusting coolant flow to prevent fuel damage, while the pressurizer system maintains pressure levels within the reactor coolant system, monitoring reactivity parameters to ensure stable operation. *HPCI* provides additional coolant injection to mitigate temperature peaks and maintain core integrity.

5. Conclusions

Modeling within *PCTRAN* involves intricate equations and models capturing the behavior of various plant components under extreme conditions. From reactor kinetics to heat transfer and emergency cooling systems, *PCTRAN* encompasses a wide array of models essential for simulating *SGTR* and *AC* power loss scenarios. These models, combined with safety criteria, boundary conditions, and initial conditions, form the basis for accurate simulations and subsequent safety assessments.

The results and discussions presented demonstrate the dynamic behavior of reactor systems under different scenarios, emphasizing the critical role of emergency core cooling and high-pressure coolant injection systems in maintaining reactor safety. Additionally, analyses of reactor parameters such as reactivity, thermal power, and coolant flow provide valuable insights into system performance and response to abnormal events. Furthermore, the analysis of radiation levels and isotopic release underscores the importance of monitoring and mitigating radioactive material release during nuclear incidents. Isotopes like *Iodine*-131, with its short half-life and high source term activity, pose significant risks to public health and necessitate effective emergency response measures.

The study has several limitations that could be addressed in future research. First, while the current study focuses on simulating specific transient scenarios such as *SGTR* and *AC* power loss, it does not account for other complex events such as loss of coolant accidents, feedwater line breaks, or seismic events, which could provide a more comprehensive understanding of nuclear power plant behavior under extreme conditions. Moreover, the simulations are based on specific initial conditions and modeling assumptions, which could affect the generalizability of the findings to other reactor designs or configurations. Future research could explore the inclusion of more diverse transient events and varying initial conditions to broaden the applicability of the results. Additionally, the study relies on *PCTRAN* simulation software, which, while robust, has its own inherent limitations in accurately replicating real-world reactor dynamics. Future research could benefit from employing a combination of different simulation tools and validation against experimental or historical accident data to enhance the accuracy and reliability of the predictions. Finally, the current study does not extensively explore the potential long-term environmental and public health impacts of radioactive releases under the combined scenarios. Future studies could integrate advanced environmental impact models to better understand these effects over extended periods.

The implication of the study is in its contribution to the ongoing efforts to ensure the safety and reliability of *VVER*-1200 reactors, which represent a significant advancement in nuclear power generation. Despite the advancements, continuous analysis and enhancement of safety systems are recognized as crucial for the reliable operation of *NPPs*. By delving into complex modeling and analysis techniques to accurately simulate combined scenarios involving *SGTR* and *AC* power loss scenarios, the study addresses a critical aspect of nuclear reactor safety.

Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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