Impact of LOCA, SG Tube Rupture, and Power Loss on VVER-1200 Reactor Safety

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Abstract:

This study analyzes the transient behavior of the VVER-1200 nuclear power plant under various failure scenarios using the PCTran VVER-1200 simulation tool. Thermal-hydraulic and safety characteristics assessed in under 600 seconds. The pressurizer, reactor building water levels, steam generators (SG-A and SG-B), reactor core, and other components are essential to the system. Systematic temperature measurements are taken at the maximum fuel, average RCS, peak clad, and reactor building levels. The talk covers fuel Doppler feedback, soluble boron's effect on reactivity, nuclear boiling ratio (DNBR) aberrations, and temperature changes. After 5 seconds, faults may develop, and system specifications affect basic circumstances. A 75% loss-of-coolant accident (LOCA) in the cold leg, a 60% power loss in the AC system, steam generator tube ruptures at 30% in SG-A and 45% in SG-B, and one auxiliary pump failure are simulated failures. The analysis found significant disparities between nuclear power station water levels and RCS leak mass flow rate. Core thermal power, nuclear flux, and turbine load changes indicate reactor performance. The results show the system's sensitivity to catastrophic events and how passive safety mechanisms stabilize the reactor. Specialists' rigorous investigation of historical accident processes improved VVER-1200 reactor safety.

Keywords: PCTran VVER-1200, LOCA, reactor safety, transient analysis, nuclear power plant, accident simulation

1.0 Introduction:

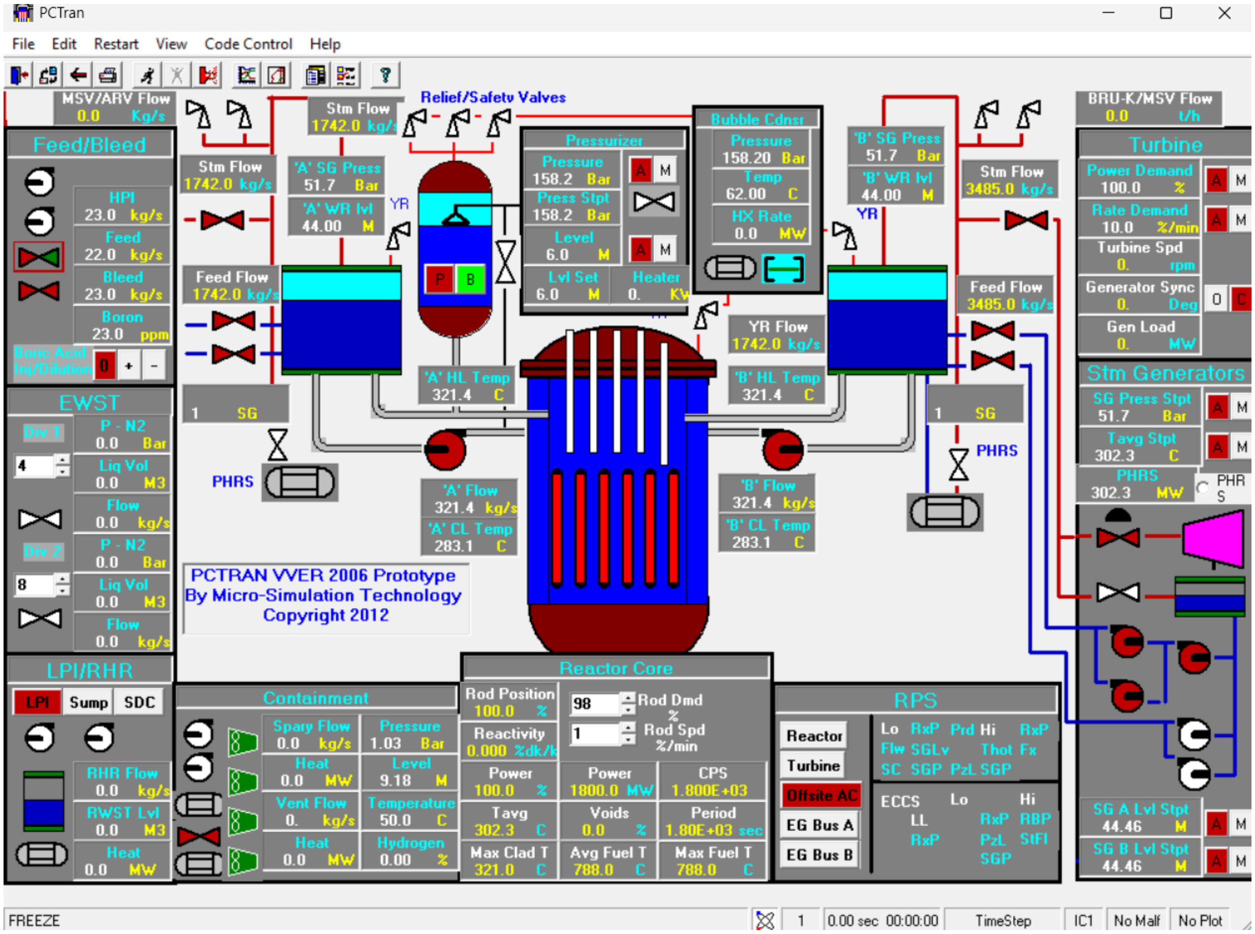
Bangladesh, classified as a developing nation, is currently facing a significant electrical energy crisis. Several natural conventional resources are present in Bangladesh. Eventually, just these energy sources will be used to generate power. Nuclear energy, often perceived as a destructive force, possesses the potential for applications in power generation and life-saving interventions. In 2009, a formal agreement was established between Bangladesh and Russia to construct the Rooppur Nuclear Power Plant (RNPP) in proximity to the Pabna district, with Rosatom designated as the leading entity for the project [1]. This research aims to offer accident analysis guidelines for NPPs with PWRs, considering their unique design aspects. This advice lists starting points in depth. Explore the causes of occurrences and their safety implications, including probable deterioration of barriers against radioactive material leakage. The focus is on licensing-type safety evaluations to demonstrate appropriate safety margins. Event analysis methodologies vary mostly in acceptance criteria and needed conservatism. Acceptance criterion examples are also supplied. Conservative assumptions, resulting in dismal findings, are suggested for typical beginning values and boundary circumstances. Guidelines for selecting acceptance criteria, beginning conditions, and boundary conditions are offered. Methodological guidance for analyzing individual incidents is provided. Suggested output parameter lists for various events. Fyza et al. [2] evaluates a severe accident in VVER-1200 during a Loss of Coolant Accident (LOCA) with loss of offsite power. It analyzes pressure, temperature, and void fraction changes, showing effective safety measures and compliance with Preliminary Safety Assessment Report (PSAR) data. Hasan Tanim et al. [3] found that the Generation III+ VVER-1200 nuclear reactor is among the safest globally. As an electrical system, it can fail. Understanding fault scenarios is essential for prevention. This study uses PCTRAN VVER-1200 (2006) prototype software to model reactor faults. Loss of feedwater flow, steam generator tube rupture, and coolant breakage are faults. Transient behavior analysis examines their impacts on reactivity, steam flow, feedwater flow, and pressure to assess reactor safety under anomalous situations. Cilliers et al. [4] emphasized the importance of early defect diagnosis in nuclear power facilities to enhance maintenance planning. The comparison of plant data with reliable references facilitated the identification of transient issues. The correlation of fault data enables operators to ascertain the location and magnitude of faults in complex scenarios, as indicated by their findings. Pawluczyk et al. [5] describe the 900MWe Westinghouse PWR reactor's Loss of Coolant Accident (LOCA) simulation approach.RELAP5 and CATHARE 2 thermal-hydraulic system codes modeled steady state and transient development. This research analyzes a 6-inch cold leg break using both codes and compares the results. undertaken during RELAP5 and CATHARE 2 benchmarking. Received bot steady-state and transient findings. Edeoza et al. [6] conducted an analysis of pressurized water reactor (PWR) safety and transients through the utilization of the PCTRAN simulator, which was employed to simulate a range of accident scenarios. The study underscored the significance of coolant pressure drops, the implications of turbine trip events, and incidents related to fuel handling, thereby illustrating the efficacy of PCTRAN in evaluating the safety of nuclear facilities, which is essential for the advancement of Nigeria's nuclear energy initiatives. Using PCTRAN, Saha et al. [7] simulated a 100% single-tube rupture in the VVER-1200 reactor. The built-in safety safeguards prevented any impact on the reactor's power or neutron flux, and the results indicated only minimal variations in pressure and temperature. The findings are consistent with PSAR data, demonstrating that the reactor is capable of reducing the impact of SGTR incidents.

This investigation examines the thermal and reactivity characteristics of a nuclear reactor system as it evolves over time. This research examines critical parameters including the departure from nuclear boiling ratio (DNBR), alterations in reactivity, and fluctuations in temperature across various reactor components. Graphical representations of data facilitate the comprehension of the evolution of these parameters throughout reactor operation and under transient conditions. The results contribute to the evaluation of the reactor's safety, operational efficiency, and its ability to respond to abrupt variations. Comprehending these behaviors is crucial for ensuring stable reactor operations and mitigating the risk of accidents. The report presents a comprehensive examination of the findings and their implications for reactor performance and safety assessment.

1.1 Illustration of PCTran:

PCTRAN, a desktop simulation program, is useful for designs of light water nuclear reactors. This system has been effectively designed for PWR, BWR, advanced AREVA EPR, Westinghouse

AP1000, GE ABWR, and ESBWR plants. PCTRAN is a program that mimics big accidents. BWR5 and Rad Puff distribute dosages from these sources. This method was developed by Micro-Simulation in 1985. PCTRAN is the teaching platform for the IAEA's yearly Simulators Workshop. Nuclear power facilities and research organizations throughout the world use plant-specific models for training, analysis, probabilistic safety evaluations, and emergency planning [7].



MSV/ARV Flow 0.0 Kg/s
Stm Flow 1742.0 kg/s
BRU-K/MSV Flow 0.0 t/h
HPI 23.0 kg/s
Feed 22.0 kg/s
Bleed 23.0 kg/s
Boron 23.0 ppm
Boric Acid Inj/Dilution 0
Stm Flow 1742.0 kg/s
'A' SG Press 51.7 Bar
'A' WR lvl 44.00 M
Pressurizer Pressure 158.2 Bar
Press Stpt 158.2 Bar
Level 6.0 M
Lvl Set 6.0 M
Heater 0. KW
Bubble Cdnsr Pressure 158.20 Bar
Temp 62.00 C
HX Rate 0.0 MW
'B' SG Press 51.7 Bar
'B' WR lvl 44.00 M
Stm Flow 3485.0 kg/s
Feed Flow 1742.0 kg/s
Feed Flow 3485.0 kg/s
YR Flow 1742.0 kg/s
'A' HL Temp 321.4 C
'B' HL Temp 321.4 C
1 SG
1 SG
'A' Flow 321.4 kg/s
'A' CL Temp 283.1 C
'B' Flow 321.4 kg/s
'B' CL Temp 283.1 C
Turbine Power Demand 100.0 %
Rate Demand 10.0 %/min
Turbine Spd 0. rpm
Generator Sync 0. Deg
Gen Load 0. MW
SG Press Stpt 51.7 Bar
Tavg Stpt 302.3 C
PHRS 302.3 MW
SG A Lvl Stpt 44.46 M
SG B Lvl Stpt 44.46 M
EWST Div 1 4
P - N2 0.0 Bar
Liq Vol 0.0 M3
Flow 0.0 kg/s
Div 2 8
P - N2 0.0 Bar
Liq Vol 0.0 M3
Flow 0.0 kg/s
RHR Flow 0.0 kg/s
RWST Lvl 0.0 M3
Heat 0.0 MW

PCTRAN VVER 2006 Prototype
By Micro-Simulation Technology
Copyright 2012

Containment Spary Flow 0.0 kg/s
Pressure 1.03 Bar
Heat 0.0 MW
Level 9.18 M
Vent Flow 0. kg/s
Temperature 50.0 C
Heat 0.0 MW
Hydrogen 0.00 %
Rod Position 100.0 %
Rod Dmd 98 %
Rod Spd 1 %/min
Reactivity 0.000 %dk/k
Power 100.0 %
Power 1800.0 MW
CPS 1.800E+03
Tavg 302.3 C
Voids 0.0 %
Period 1.80E+03 sec
Max Clad T 321.0 C
Avg Fuel T 788.0 C
Max Fuel T 788.0 C

FREEZE | 1 | 0.00 sec 00:00:00 | TimeStep | IC1 | No Malf | No Plot

Fig.1. The visual user interface of PCTRAN

2.0 Initial Condition and Malfunction Setup:

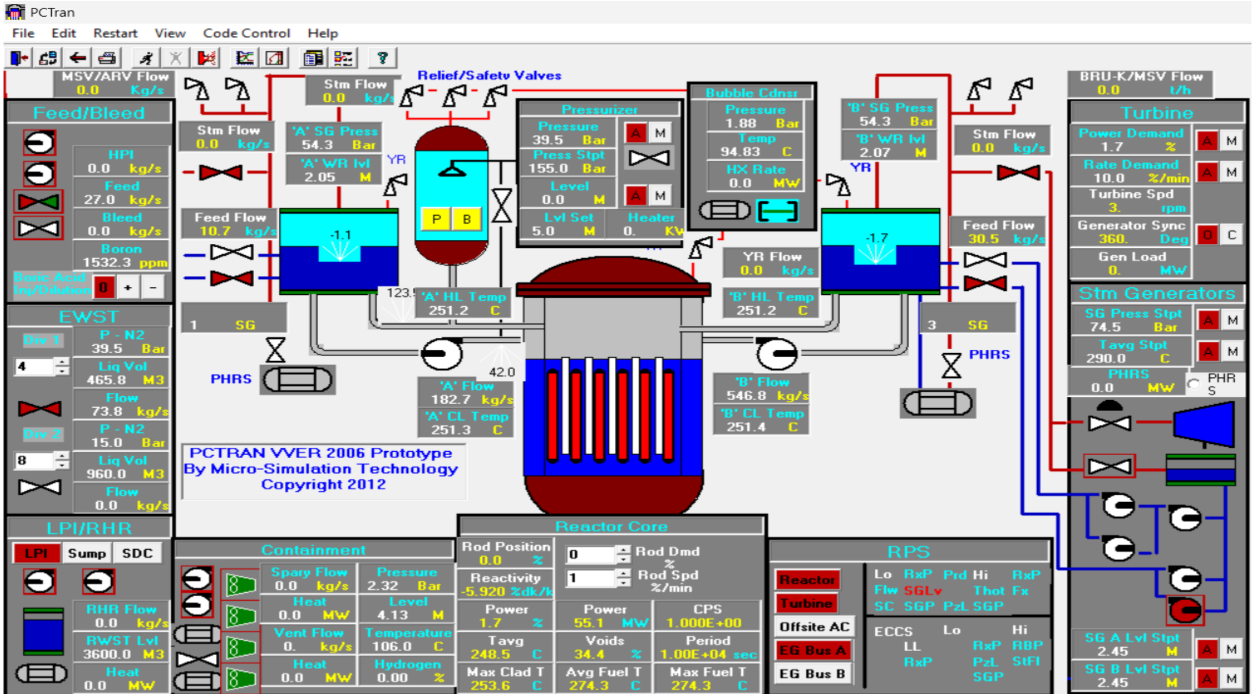
The simulation was carried out using IC Set No. 7, which had a starting power of 1.94 MW, a reactor coolant pressure of 154.63 bar, an average temperature of 289.48 K, and a steam generator pressure of 71.26 bar. These conditions were all measured at the beginning of the cycle (BOC) when the system was shut down. Five seconds after the simulation began, malfunctions were added. These included a 75% failure fraction in the cold leg Loss of Coolant Accident (LOCA), a 60% loss of AC power, tube ruptures in steam generators A and B at 30% and 45%, respectively, and the failure of one auxiliary pump. An analysis was conducted on the system's response to these failures

Table 1: Initial Conditions (IC Set No.7)

Parameter	Value
Power	1.94 MW
RC Press	154.63 bar
Tavg	289.48 K
SG Press	71.26 bar
Time in Life	BOC (Beginning of Cycle)
Description	Shutdown

Table 2: Malfunction Scenarios

Malfunction Type	Failure Fraction /Condition
LOCA in Cold Leg	75% Failure Fraction
Loss of AC Power	60%
Tube Rupture (Generator A)	30%
Tube Rupture (Generator B)	45%
Auxiliary Pump Malfunction	One Pump Affected



MSV/ARV Flow 0.0 Kg/s
Stm Flow 0.0 kg/s
Stm Flow 0.0 kg/s
'A' SG Press 54.3 Bar
'A' WR lvl 2.05 M
Feed Flow 10.7 kg/s
HPI 0.0 kg/s
Feed 27.0 kg/s
Bleed 0.0 kg/s
Boron 1532.3 ppm
P - N2 39.5 Bar
Liq Vol 465.8 M3
Flow 73.8 kg/s
P - N2 15.0 Bar
Liq Vol 960.0 M3
Flow 0.0 kg/s
RHR Flow 0.0 kg/s
RWST Lvl 3600.0 M3
Heat 0.0 MW
Spary Flow 0.0 kg/s
Pressure 2.32 Bar
Heat 0.0 MW
Level 4.13 M
Vent Flow 0. kg/s
Temperature 106.0 C
Heat 0.0 MW
Hydrogen 0.00 %
Pressurizer Pressure 39.5 Bar
Press Stpt 155.0 Bar
Level 0.0 M
Lvl Set 5.0 M
Heater 0. KW
Bubble Cdnsr Pressure 1.88 Bar
Temp 94.83 C
HX Rate 0.0 MW
YR Flow 0.0 kg/s
'A' HL Temp 251.2 C
'B' HL Temp 251.2 C
'A' Flow 182.7 kg/s
'A' CL Temp 251.3 C
'B' Flow 546.8 kg/s
'B' CL Temp 251.4 C
'B' SG Press 54.3 Bar
'B' WR lvl 2.07 M
Stm Flow 0.0 kg/s
Feed Flow 30.5 kg/s

PCTRAN VVER 2006 Prototype
By Micro-Simulation Technology
Copyright 2012

Rod Position 0.0 %
Rod Dmd 0 %
Rod Spd 1 %/min
Reactivity -5.920 %dk/k
Power 1.7 %
Power 55.1 MW
CPS 1.000E+00
Tavg 248.5 C
Voids 34.4 %
Period 1.00E+04 sec
Max Clad T 253.6 C
Avg Fuel T 274.3 C
Max Fuel T 274.3 C
BRU-K/MSV Flow 0.0 t/h
Power Demand 1.7 %
Rate Demand 10.0 %/min
Turbine Spd 3. rpm
Generator Sync 360. Deg
Gen Load 0. MW
SG Press Stpt 74.5 Bar
Tavg Stpt 290.0 C
PHRS 0.0 MW
SG A Lvl Stpt 2.45 M
SG B Lvl Stpt 2.45 M

Fig.2. Visualization of PCTran after 600 sec runs with the IC condition and being Malfunctione

3.0 Results and Discussion:

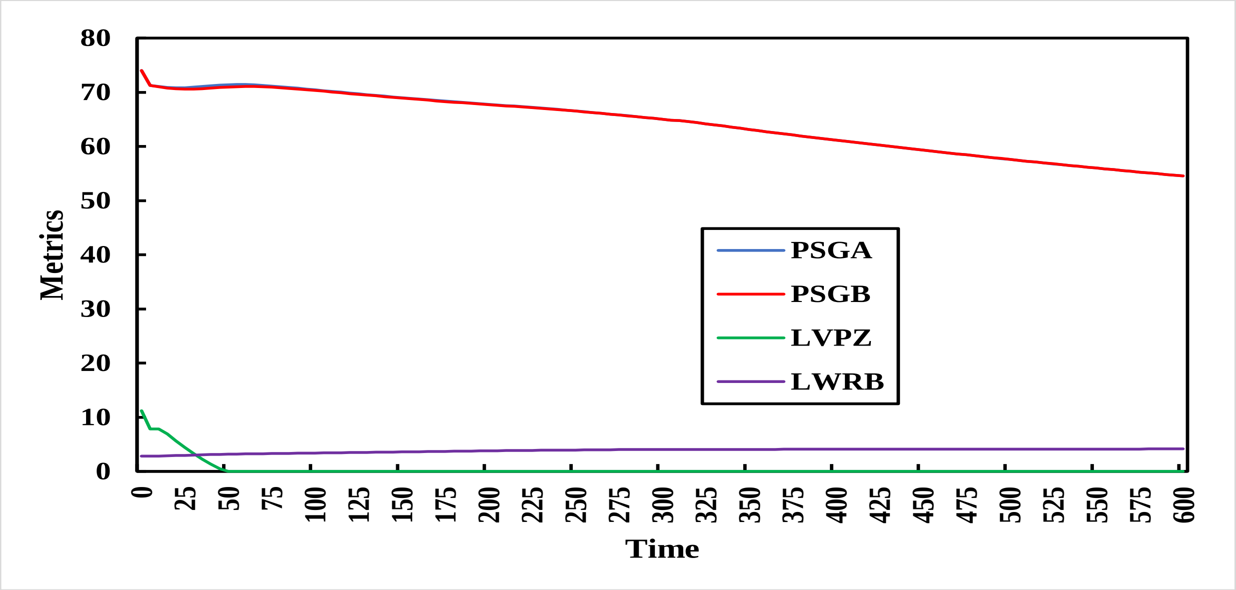


Fig.3. Different Variable (PSGA, PSGB, LVPZ, LWRB) against Time

Pressure Steam Generator A (PSGA) and Pressure Steam Generator B (PSGB) fluctuations over a specific time scale are shown in Figure 3, together with the Level Pressurizer (LVPZ) % and the Level Reactor Building Sump Water (LWRB) meters. The PSGB (red line) starts at about 72 bar and gradually drops down over the time period that was monitored. As seen by the blue line, the PSGA shows a trend that is quite consistent across all data sets, with only small deviations. The green line, which represents the LVPZ, shows a sharp drop at the outset before quickly leveling off near zero. Throughout the interval that was examined, the LWRB (purple line) shows a consistent and gradual rise. Among the metrics, the data show a notable tendency toward rising sump water levels and fluctuating steam generator pressures.

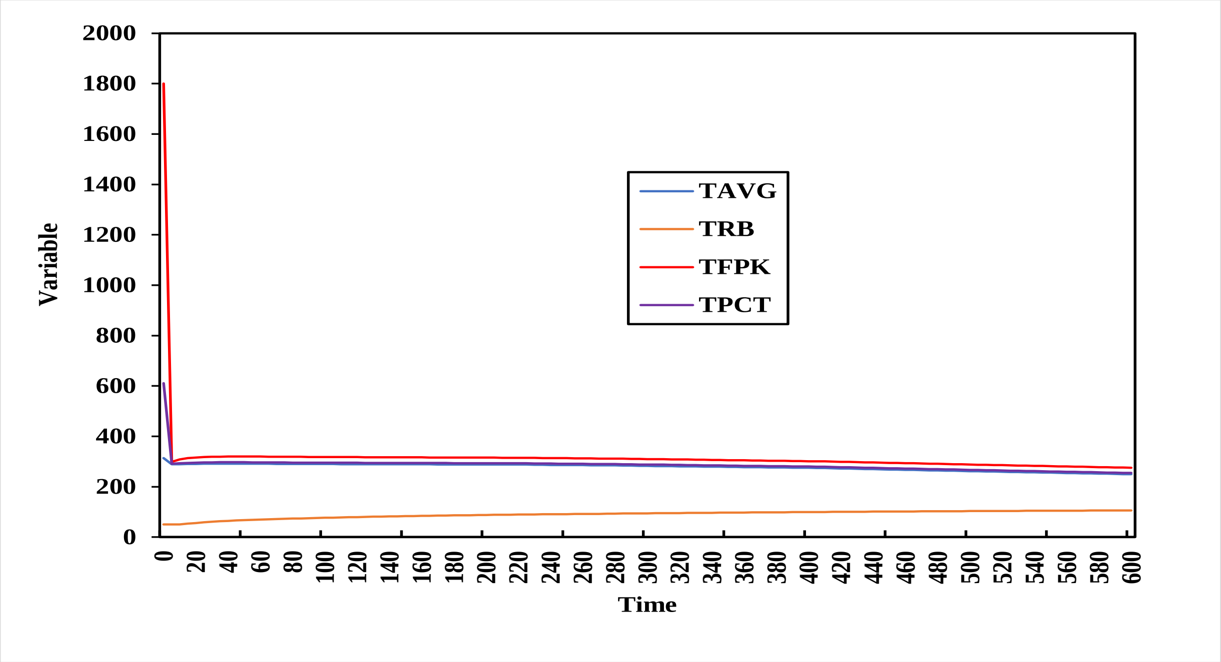


Fig.4. Different variable (TAVG, TRB, TFPK, TPCT) against Time

TAVG (Temperature RCS Average in °C), TRB (Temperature Reactor Building), TFPK (Temperature Peak Fuel), and TPCT (Temperature Peak Clad) vary over 600 seconds in Fig.4. TFPK (red) first peaks over 1800°C, then rapidly decreases before settling at 250-300°C. TAVG (blue) and TPCT (purple) have similar patterns, although they consistently register values somewhat lower than TFPK, demonstrating their thermal sensitivity to reactor transients. The TRB (orange) starts at a low temperature and climbs, showing persistent heat transfer from the reactor core to the neighboring structure. The first spike in TFPK and TPCT suggests a reactor event, maybe reactivity insertion or a transient power surge, followed by heat dissipation stability. The controlled performance of TAVG and TRB supports effective system cooling and heat transmission.

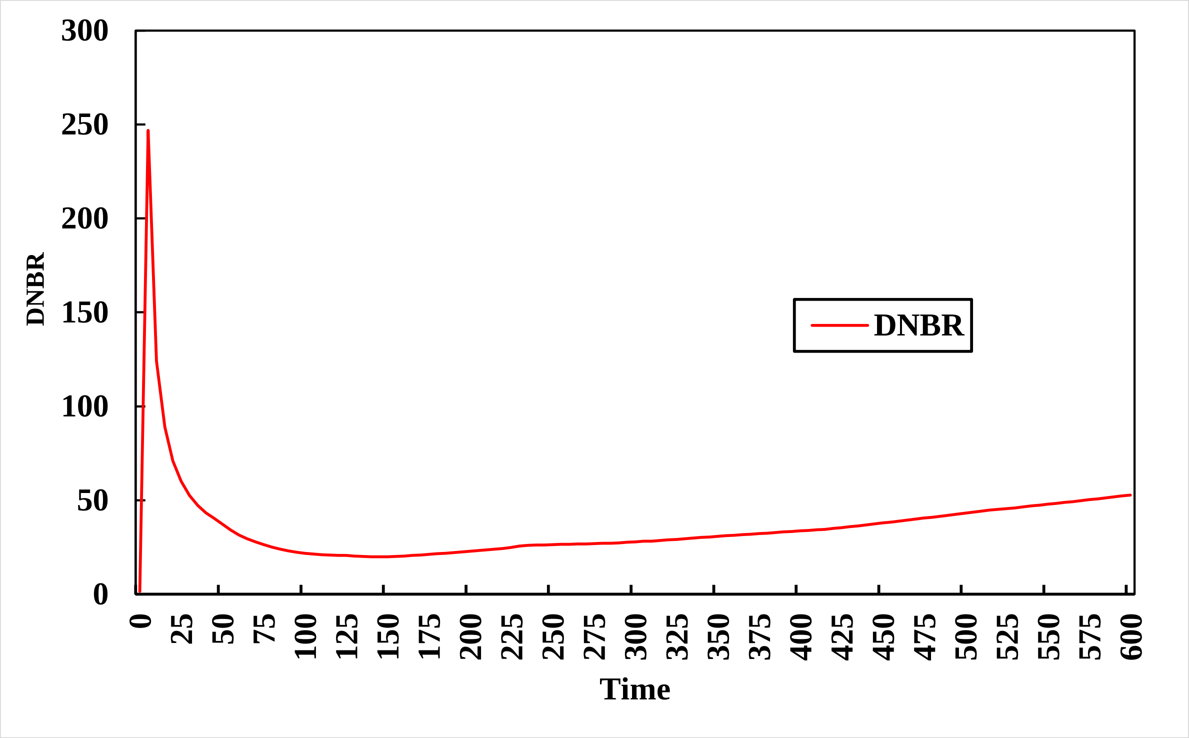


Fig.5. DNBR against Time

One important safety metric that displays heat flow margin is the Departure from Nuclear Boiling Ratio (DNBR), and Fig. 5 depicts its volatility across a 600-second simulation. At first, DNBR could spike abruptly to around 250 due to a sudden drop in pressure or change in coolant flow. It falls below 50 within 50 seconds, which means that heat transfer is reduced. It appears that DNBR stabilizes at a lower value after 150 seconds and gradually increases after 300 seconds, indicating a progressive recovery. This might be because to emergency cooling activation or a drop in reactor output. Reactor safety is guaranteed by DNBR values between 1.3 and 1.5, whereas fuel overheating danger increases with lower values. A thermal-hydraulic temporary and corrective response is implied by a sharp decline and stability. Reactor cooling, fuel damage resistance, and DNBR slow growth are all improved at the end.

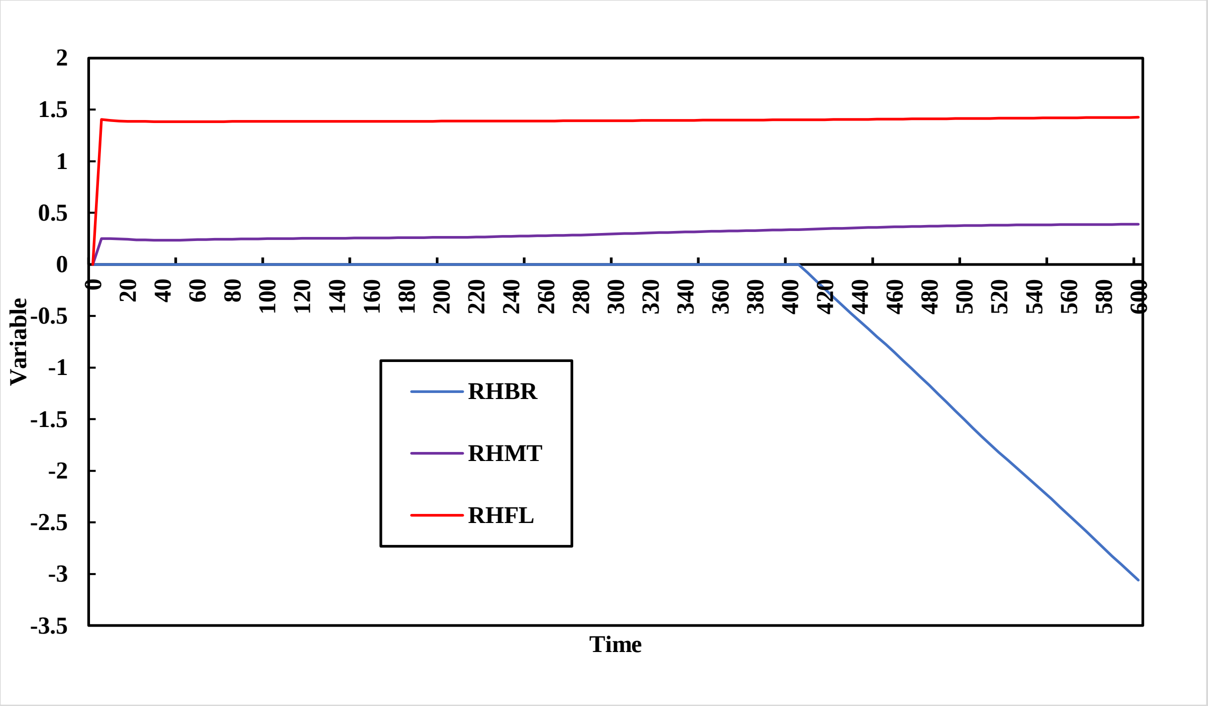


Fig.6. Different Variable (RHBR, RHMT, RHFL) against Time

Over the course of 600 seconds, the %dk/k fluctuations of RHBR (Reactivity Soluble Boron), RHMT (Reactivity Mod Temperature), and RHFL (Reactivity Fuel or Doppler) are shown in Figure 6. The blue RHBR begins at a near-zero value but rapidly decreases after 420 seconds, reaching -3.5% dk/k at the conclusion. This indicates that fluctuations in boron concentration are producing a significant negative reactivity insertion. Observing the little upward trend in RHMT (purple) indicates a tiny but growing positive temperature feedback. The red Doppler reactivity curve (RHFL) reaches 1.5% dk/k in a matter of seconds and then stays there, indicating that there is a dominant negative feedback mechanism preventing power escalation. A larger infusion of boron may be necessary for reactivity control beyond 420 seconds, as a result of the significant fall in RHBR. Reactor stability is achieved by a balance of reactivity, as shown graphically by Doppler and temperature effects.

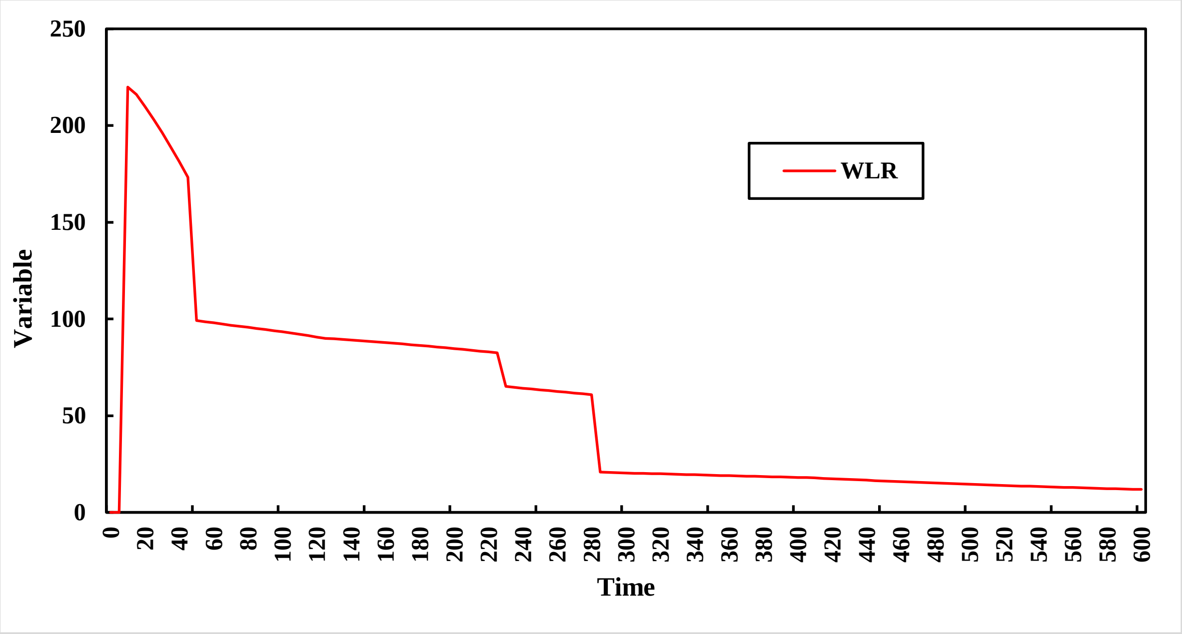


Fig.7. WLR against Time

Figure 7 illustrates how WLR (Flow RCS Leak) varies during a simulated period of 600 seconds, measured in kg/s. In the beginning, WLR rises quickly, reaching a peak of around 220 kg/s in the first few seconds due to the simulated LOCA in the cold leg. A reduction is then demonstrated, with multiple dramatic drops occurring in connection with significant events such as an increase in system failures and power outages. The WLR decreases below 100 kg/s and continues to diminish steadily after approximately 260 seconds. At the end of the simulation, the leakage stabilizes at a low value. This indicates that either the coolant inventory has run out or the system has found equilibrium following a failure.

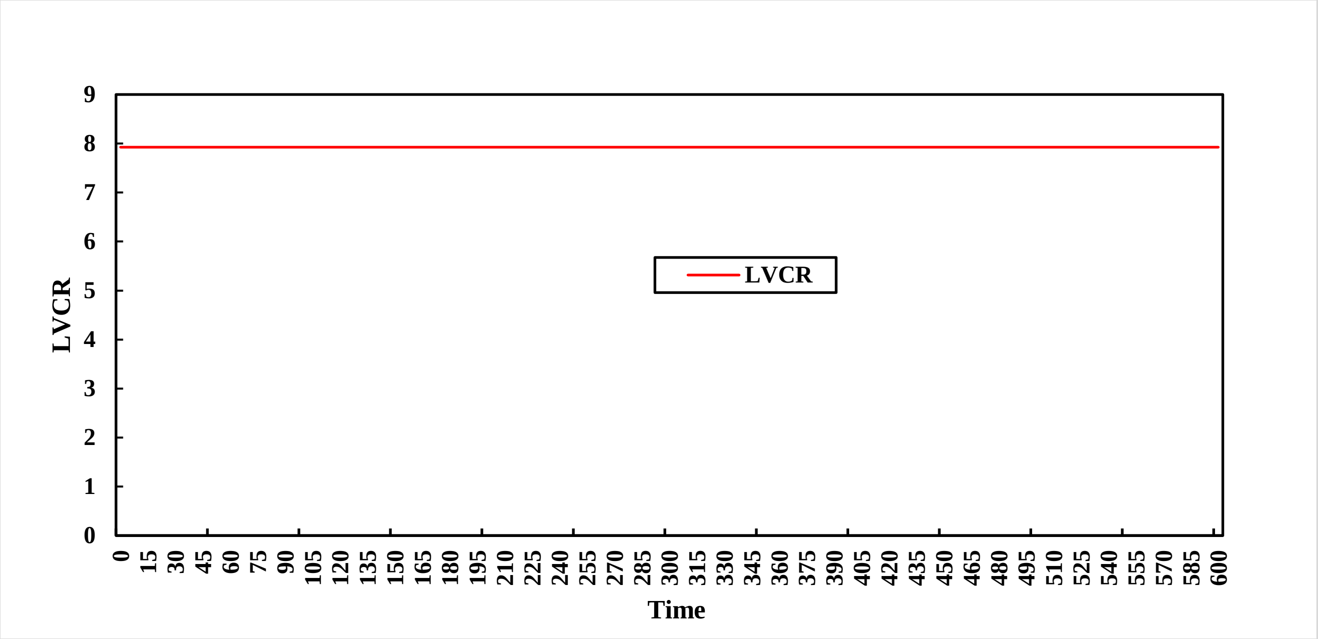


Fig.8. LVCR against Time

Figure 8, which displays the Level Core Water (LVCR) across a time interval of 1 to 600 seconds, seems to be a horizontal line at around 7.9249 meters. The core water level would appear to be constant and unchanging on the graph as the LVCR readings remain constant over the whole duration. This may be because the nuclear reactor is operating in a steady-state condition or because the control systems are very good at maintaining a consistent water level in the core.

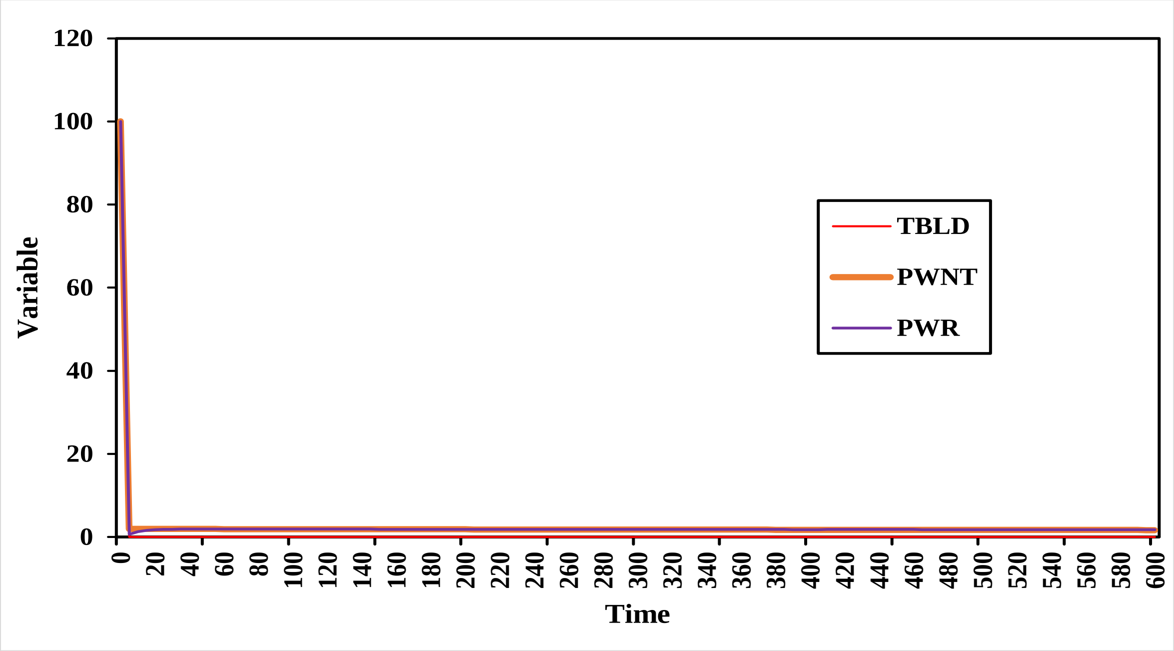


Fig.9. Different Variable (TBLD, PWNT, PWR) against Time

An example of the time-varying components of a nuclear power plant's power output is shown in Figure 9, which stands for power turbine load, power core thermal, and power nuclear flux. With TBLD (red line), PWNT (orange line), and PWR (purple line) all reaching almost 100% in the beginning, all three variables see a tremendous surge. Rapid deterioration follows shortly thereafter, with all parameters falling sharply and remaining around zero for the rest of the time. According to this pattern, everything started out with a lot of power and then quickly went downhill. This may mean that the turbine broke down, the reactor was shut down, or that an emergency reaction scenario occurred, meaning that power output was low for a lengthy time.

4.0 Conclusion:

This research examines critical accident scenarios within the VVER-1200 nuclear reactor framework, employing the PCTRAN simulator under a specified initial condition (IC Set No. 7). The simulation, conducted over a duration of 600 seconds, investigated the implications of a Loss of Coolant Accident (LOCA) occurring in the cold leg, along with a steam generator tube rupture and an auxiliary pump failure. The findings reveal a notable decrease in reactor coolant system (RCS) pressure accompanied by the activation of safety mechanisms; however, the loss of AC power resulted in the malfunction of the high-pressure injection (HPI) system. As a result, the reactor trip was initiated, effectively averting any potential escalation of the incident. In light of the variations observed in system parameters, the implemented safety features successfully reduced the potential consequences. The results are consistent with the data presented in the Preliminary Safety Assessment Report (PSAR), thereby reinforcing the resilience of the reactor's design. The findings presented here play a crucial role in improving accident management strategies and in deepening the understanding of the VVER-1200's safety response during severe transient conditions.

5.0 Major Findings:

I. LOCA caused a rapid drop in RCS pressure.
II. Reactor trip occurred during the simulation.
III. Safety features responded effectively but were impacted by AC power loss.
IV. Fuel temperature remained controlled within safe limits.
V. The HPI system activated but later failed due to power loss.
VI. Steam generator tube rupture contributed to pressure and temperature variations.
VII. Results align with PSAR data, confirming reactor safety capabilities

Data Availability: Data will be made available on request.

Nomenclature

Symbol/Abbreviation	Full Form/Description	Unit
VVER-1200	Water-Water Energetic Reactor-1200	-
LOCA	Loss of Coolant Accident	-
RCS	Reactor Coolant System	-
DNBR	Departure from Nuclear Boiling Ratio	-
RHBR	Reactivity Soluble Boron	%dk/k
RHMT	Reactivity Mod Temperature	%dk/k
RHFL	Reactivity Fuel (Doppler)	%dk/k
TAVG	Reactor Coolant System Average Temperature	°C
TRB	Reactor Building Temperature	°C
TFPK	Peak Fuel Temperature	°C
TPCT	Peak Cladding Temperature	°C
SGTR	Steam Generator Tube Rupture	-
MW	Megawatt (Power)	MW
Bar	Pressure Unit	Bar
BOC	Beginning of Cycle	-
SG Press	Steam Generator Pressure	Bar
TimelnLife	Time in Reactor Core Life	-
RCS	Reactor Coolant System	-

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