**Royal Institute of Technology**
Department of Nuclear Engineering

# Compact Reactor Simulator Transient Exercises with Apros 6

Aurélien Pierre Morel
Federico Pati
David Ploumhans
Wilhelm Ecker

February 11, 2026

Professor: Sean Roshan Ghias
Course code: SH2705

# Contents

# 1 Introduction

This report presents a series of transient simulations performed on a Boiling Water Reactor (BWR) using the APROS 6 modeling environment. The simulations aim to analyze the reactor's dynamic behavior under various perturbation scenarios, ranging from feedwater system failures to emergency shutdowns. The insights gathered from these cases contribute to a better understanding of thermal-hydraulic feedback mechanisms and control system responses during non-steady-state conditions.
The reactor model employed is the Generic APROS 6 BWR model, featuring a one-dimensional thermal-hydraulic layout coupled with a two-group neutronics model. The core neutron kinetics are modeled with 6 delayed neutron groups, providing a realistic depiction of the reactor's behavior under transient conditions. The top of the reactor core is defined as the reference elevation (0.000 m) for all vertical positions in the simulation model.

**Steady-State Operating Conditions**

Table 1: Reactor Parameters at Steady State

| **Parameter** | **Value** |
|---|---|
| Thermal Power | 2,550 $MW_{th}$ (106% of nominal) |
| Steam Dome Pressure | 7.0 MPa |
| Steam Flow Rate | 1310 kg/s |
| Feedwater Temperature | 185.3°C |
| Recirculation Flow Rate | 7600 kg/s |
| Reactor Vessel Water Level | 4.3 m |
| Reactor Vessel Total Volume | 414.59 $m^3$ |
| Number of Recirculation Pumps | 6 |
| Feedwater Pumps | 3 + 1 (standby) |

The overall objective of this study is to understand the reactor's behavior during transient events and assess the performance of its automatic and manual control systems using detailed APROS 6 simulations. The scope of the analysis includes both thermohydraulic and neutronic responses under disturbed operating conditions.
The report is structured as follows:

- **Section I – Pressure Transient:** Investigates the pressure response with and without SCRAM signals, including the effect of SCRAM delay.
- **Section II – Steam Line Break:** Analyzes the impact on system pressure and reactor stability.
- **Section III – Feedwater Line Break:** Explores core cooling degradation and system imbalance following a line rupture.
- **Section IV – Feedwater Pump Trip:** Examines reactor behavior under complete loss of feedwater pumps, with emphasis on transient flow and temperature changes.
- **Section V – Feedwater Enthalpy Decrease:** Evaluates the reactor's thermal response to low-enthalpy feedwater and the resulting power variations.
- **Section VI – Reduced Reactor Power Effects:** Investigates system-wide behavior and control response during deliberate power reduction.

- **Section VII – Manual SCRAM:** Presents cases where emergency shutdown is initiated manually, with analysis of timing and operational response.

The following sections provide detailed descriptions, expectations, and discussions for each transient, supported by simulation data and graphical analysis.

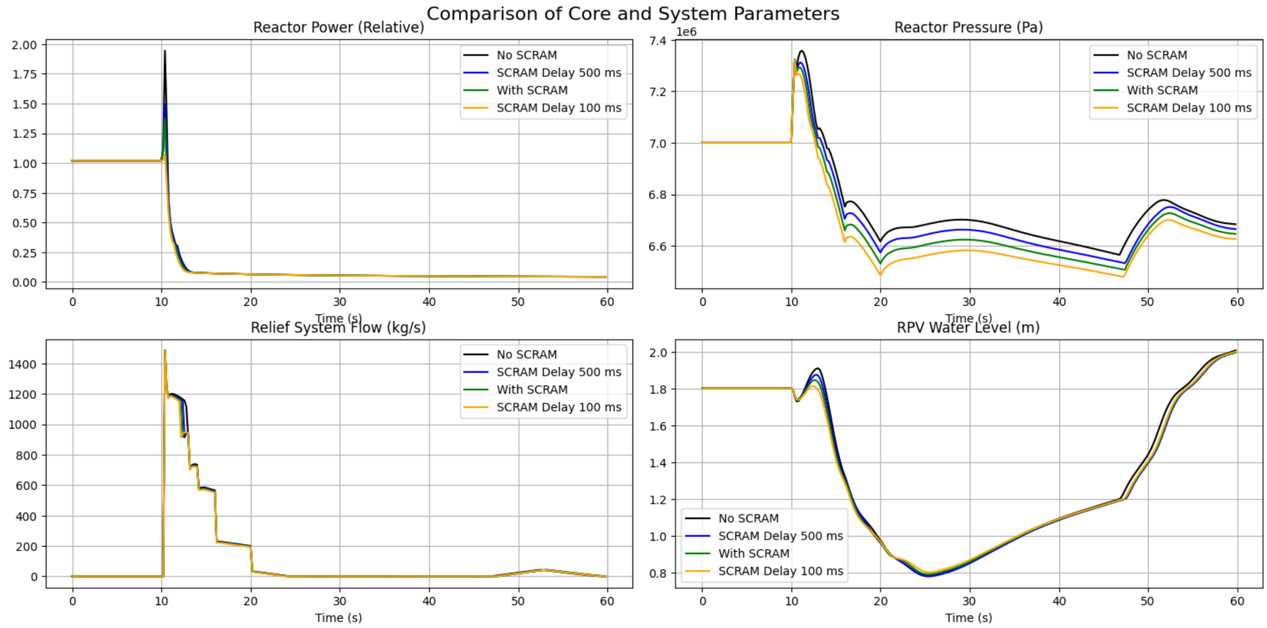


Figure 1: Comparison of deifferent delays of SCRAM

# 2 Pressure transient

This transient simulates the inadvertent closure of a main steam line (MSL), triggered by a false signal (516-A) that incorrectly indicates a steam line break. As a result, the reactor protection system initiates a series of safety actions typically associated with a steam line rupture. However, the reactor SCRAM due to 516-A is either suppressed or triggered with multiple delay values.

## 2.1 Expectation

We expect the closure of the main isolation valve to cause a rapid increase in pressure. This increase will collapse the steam bubbles in the boiling channels, reducing steam quality. The lower quality enhances moderation and thus increases reactor power. In the case where 516-A is suppressed, we expect a SCRAM signal due to either overpressure or overpower.

## 2.2 Results and Discussion

### 2.2.1 Multiple SCRAM Delays

First, it is interesting to analyze the effect of different SCRAM delays. The required activation time is 300 ms after the signal. This delay is then shortened to 100 ms, extended to 500 ms, or entirely prevented until another signal is triggered.
As shown in Figure 1, the reactivity insertion due to the reduction in void fraction may or may not be sufficient to significantly increase power. With a 100 ms delay, the power ramp is limited to less than 8%, and it is immediately suppressed by the SCRAM. On the contrary, when the SCRAM is delayed, power increases by up to 95%. Apart from this, the post-SCRAM behaviour is very similar across the cases and mainly differs by the presence of a pressure rebound caused by steam generation due to higher power levels.

### 2.2.2 SCRAM Signal 516-A Prevented

As the overall transient evolution is similar, this section focuses in more detail on the scenario where the SCRAM signal 516-A is suppressed. Despite the signal suppression, the transient still activates the following automatic safety actions:

- Closure of the inner Main Steam Line Isolation (MSLI) valves at $t = 0.1$ s.
- Closure of the outer MSLI valves at $t = 57$ s.
- Activation of the pressure relief system, both from the logic signal and the high-pressure condition in the reactor vessel.

Following the closure of the steam outlet, the steam flow rate rapidly drops to zero, leading to a sharp increase in reactor pressure. This causes a decrease in void fraction (i.e., an increase in the liquid-to-steam ratio), also observable as a drop in reactor water level.
The reduction in voids enhances moderation, leading to a temporary increase in reactor power. This power excursion triggers a SCRAM through the “high reactor power versus circulation flow (Q > 125%) – unfiltered” signal.

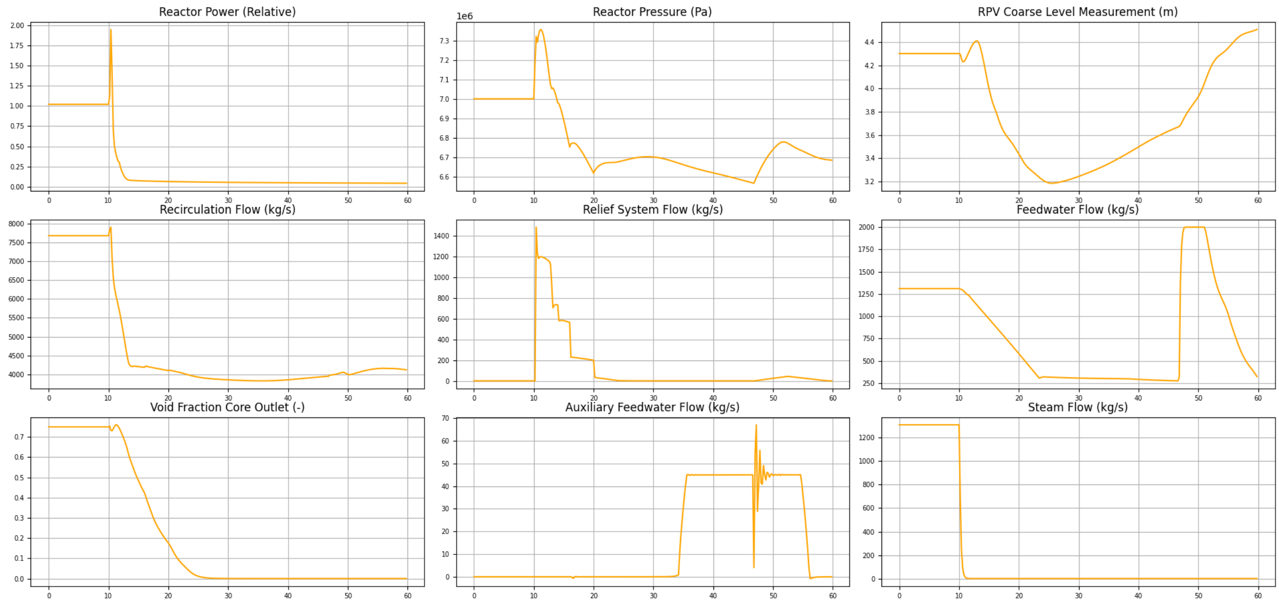


Figure 2: Key parameters evolution for the first 60 seconds.

Once the SCRAM is initiated:

- Control rods are inserted, causing a rapid drop in power up to decay heat levels.
- Boiling ceases, resulting in a steep drop in average void fraction.
- Pressure and water level fall due to the loss of boiling and the mass removal through the relief valves.

The relief system induces a characteristic sawtooth pattern in the pressure profile due to the periodic opening and closing of the valves.
From an engineering standpoint:

- Recirculation pump speed is reduced to approximately 50% of nominal to minimize power and enhance cooling.

- The feedwater system transitions to shutdown mode, with flow coasted down to around 250 kg/s.
- The Auxiliary Feedwater (AFW) system is activated with a delay of approximately 20 s.

At $t = 25$ s, the relief valves close, resulting in a slight pressure rebound. However, due to the loss of voids and continued steam venting, the water level drops below 3.6 m. After about 20 s, this condition triggers the return of the feedwater system to normal operation.
With the outlet still isolated but recirculation pumps active, the water level begins to rise (by approximately 0.5 m over 25 s). At $t = 35$ s the AFW starts, followed by the main feedwater system at $t = 50$ s, leading to a sharper increase in both pressure and water level. Once the nominal water level of 4.3 m is reached, both AFW and FW are shut off.

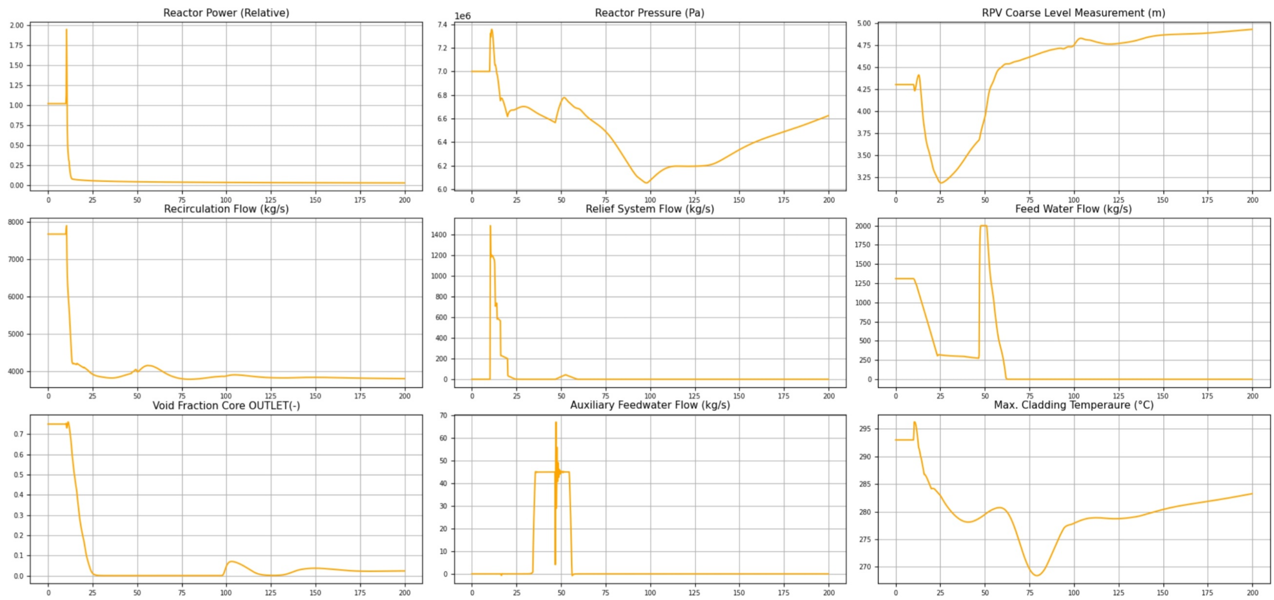


Figure 3: Key parameters evolution for the first 200 seconds.

After $t = 50$ s, pressure decreases again due to the absence of boiling (void fraction $\approx 0$ at the outlet), while the injected feedwater circulates, cools down, and condenses steam, thereby reducing pressure.
As the recirculated water heats up, temperature increases again. Around $t = 100$ s, boiling resumes and void fraction starts to rise. Consequently, pressure gradually increases and by $t \approx 300$ s exceeds the operating setpoint. This triggers the reopening of the relief valves, which maintain pressure at a stable upper level.
During this final phase, the water level decreases again due to sustained boiling and steam venting through the relief system, while no additional feedwater is injected. This leads to a pressure balance between steam generation and relief flow, causing a drop in RPV water level. When the level becomes sufficiently low, AFW restarts, initiating a cyclic, damped oscillatory behavior.

# 3 Steamline break accident

## 3.1 Expectation

We expect a loss of pressure and of steam leading to an increase in temperature of the dry and wetwell. As this accident is a LOCA, we also expect the ECCS to activate.

## 3.2 Results and Discussion

**0 to 60s summary**

We ran steady state for 10s then opened the steamline break valves to simulate a 200% guillotine break.

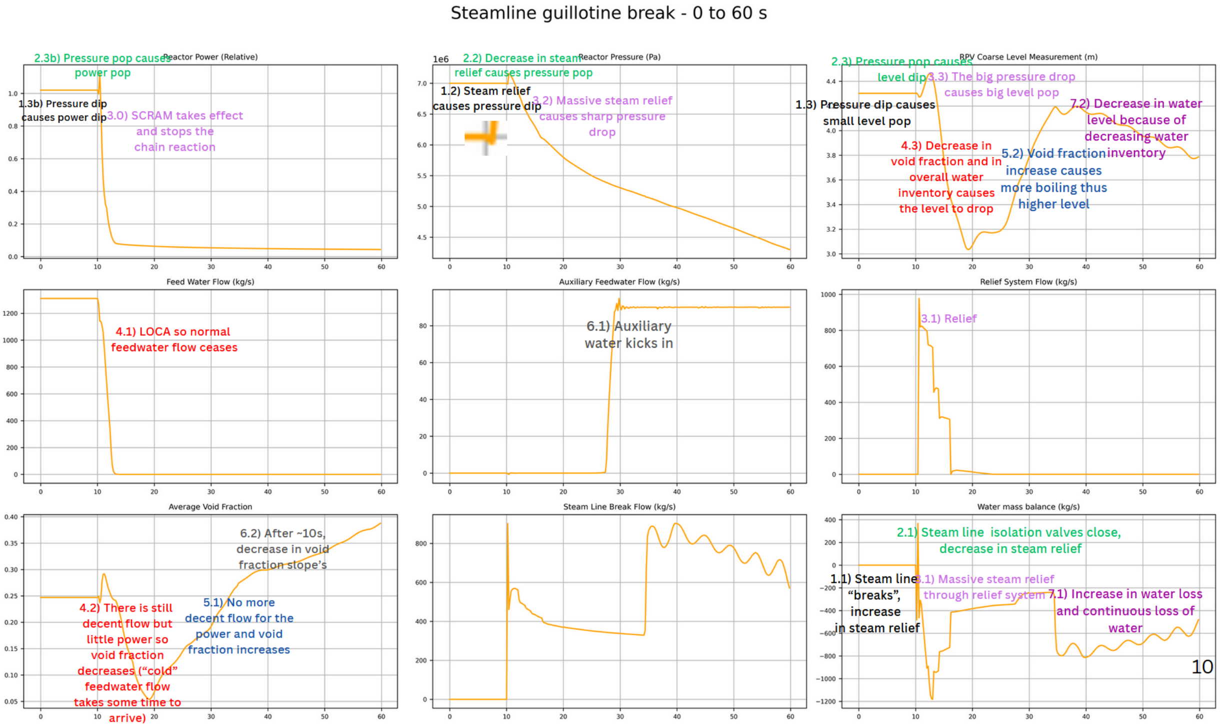


Figure 4: Visual of the key parameters evolution for the first 60 seconds of the steamline break accident.

As soon as the steam line break valves open, a SCRAM signal is sent. The control rods start rising after 300ms.

The initial little pressure dip is caused by the increased steam release (negative net water mass flow). The pressure dip causes a power dip and a water level pop. Then, the steam line isolation valves close and the relief system activates very shortly after. In between, the total release of steam decreases, with a positive net water mass flow) and the pressure pops, causing a power pop and a water level dip.

As the power and the negative net water mass flow both decrease, the pressure steadily decreases.

The combination of power drop while still having some feedwater flow (that arrives to the core with a delay) results in the decrease in the void fraction and coarse water level until the 20th second.

After, the lack of fresh water flow while still having decay heat is responsible for the rise in void fraction and of coarse water level. Then, the void fraction still increases but the overall decrease in water inventory, as more water is going in than out makes for the water level to decrease.

The auxiliary feedwater reduces the heating of the water inventory, decreasing the slope in the increase of the void fraction. The overall decrease in water inventory because of the increased steam relief through the break results in an increase in water level.

**0 to 5800s summary**

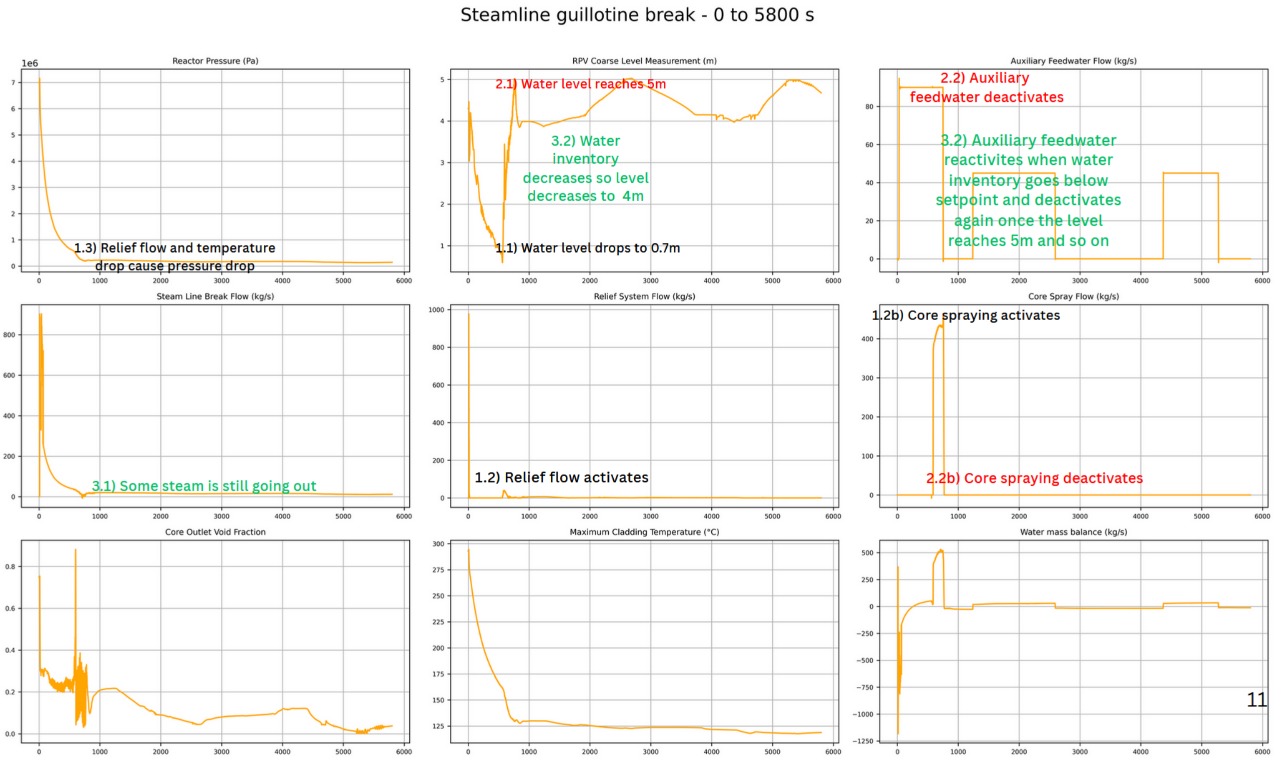


Figure 5: Visual of the key parameters evolution for the first 5800 seconds of the steamline break accident.

The coarse water level steadily decreases and upon reaching to 0.7m above core, both relief and core spraying activate, causing the pressure and temperature to dip.

The net water mass flow becomes highly positive and both water inventory and water level rise. When the water reaches 5m, both the core spray flow and the auxiliary feedwater flow deactivate.

The auxiliary feedwater will reactivate at half the flowrate after the water level falls below a set point until the level rises to 5m again and so on as there is still steam going out of the vessel through the steamline break.

**All records**

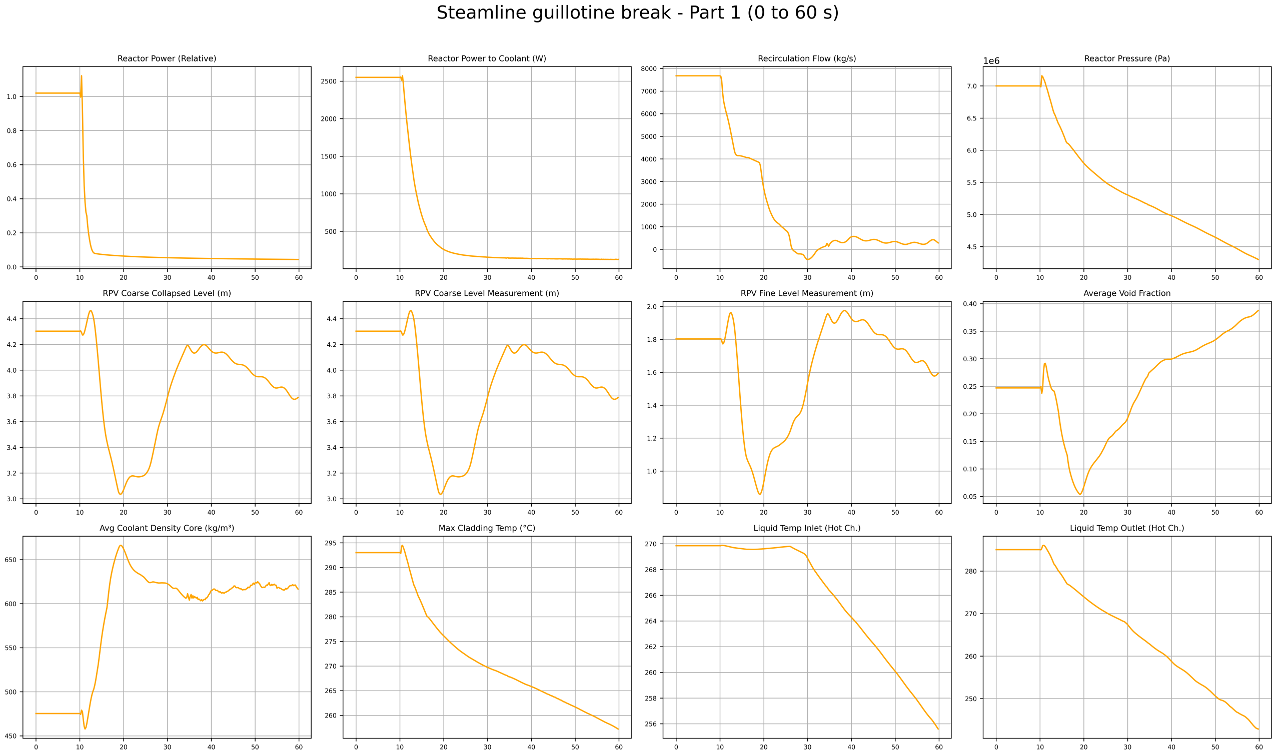


Figure 6: Visual of the 1st set of recorded parameters evolution for the first 60s of the steamline break accident.

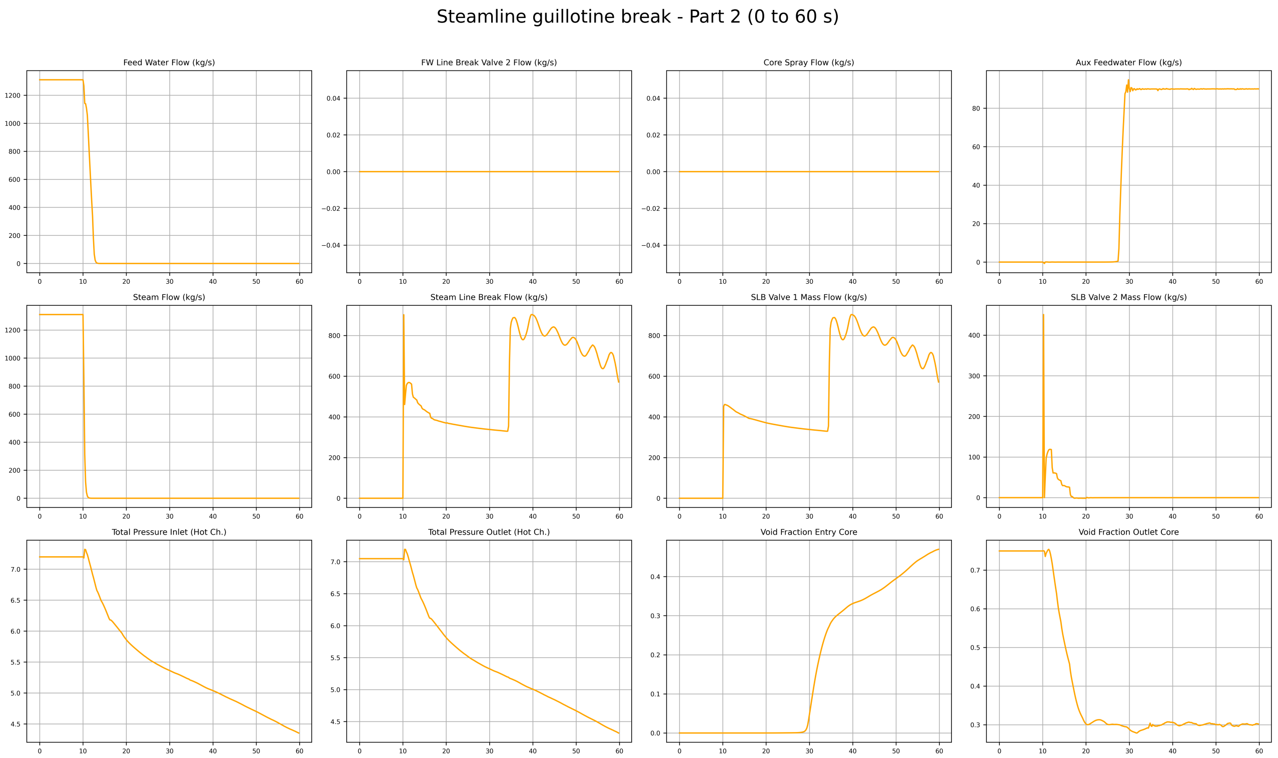


Figure 7: Visual of the 2nd set of recorded parameters evolution for the first 60s of the steamline break accident.

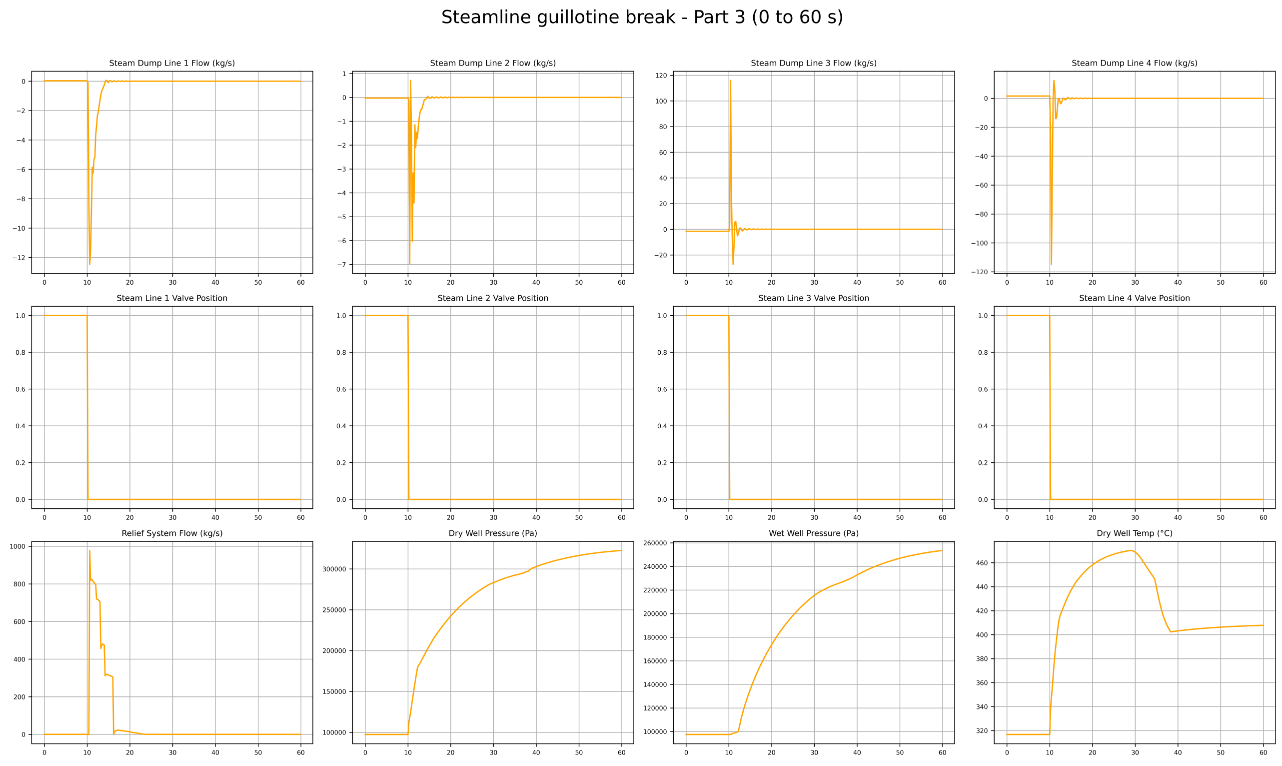


Figure 8: Visual of the of the 3rd set of recorded parameters evolution for the first 60s of the steamline break accident.

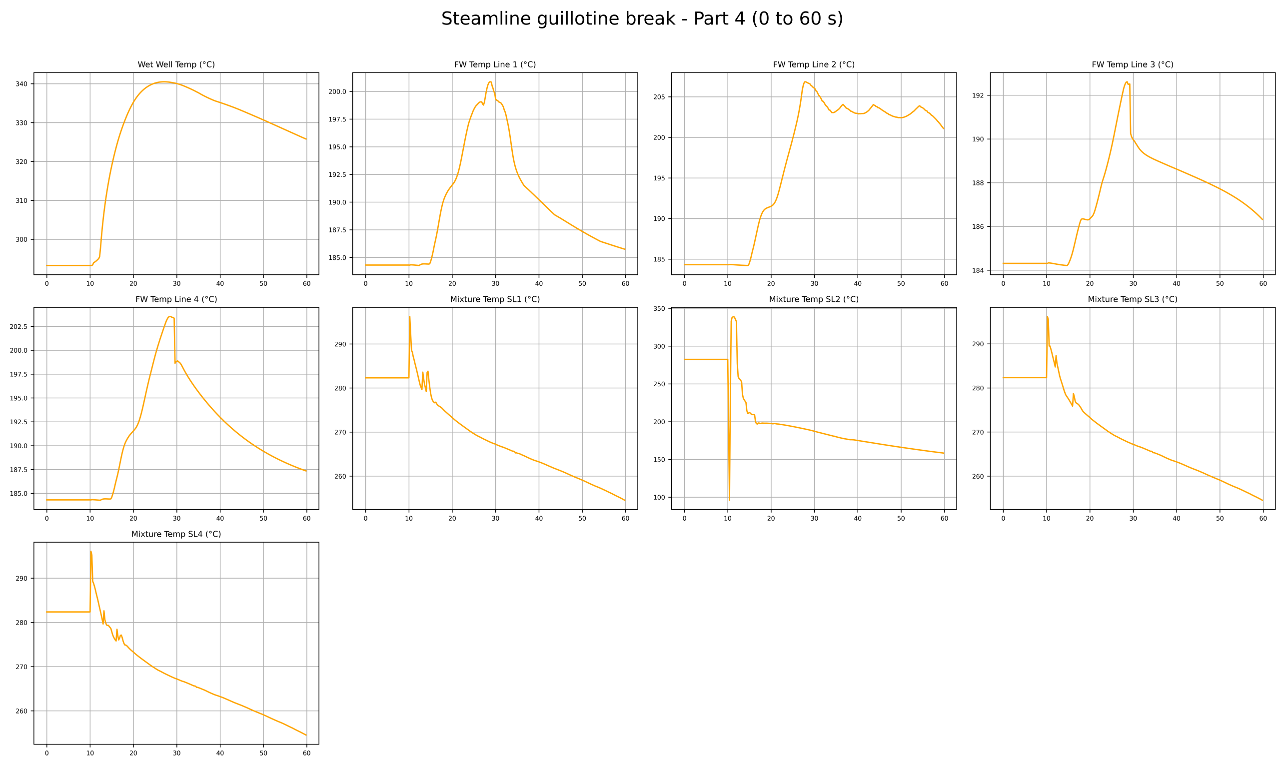


Figure 9: Visual of the 4th set of recorded parameters evolution for the first 60s of the steamline break accident.

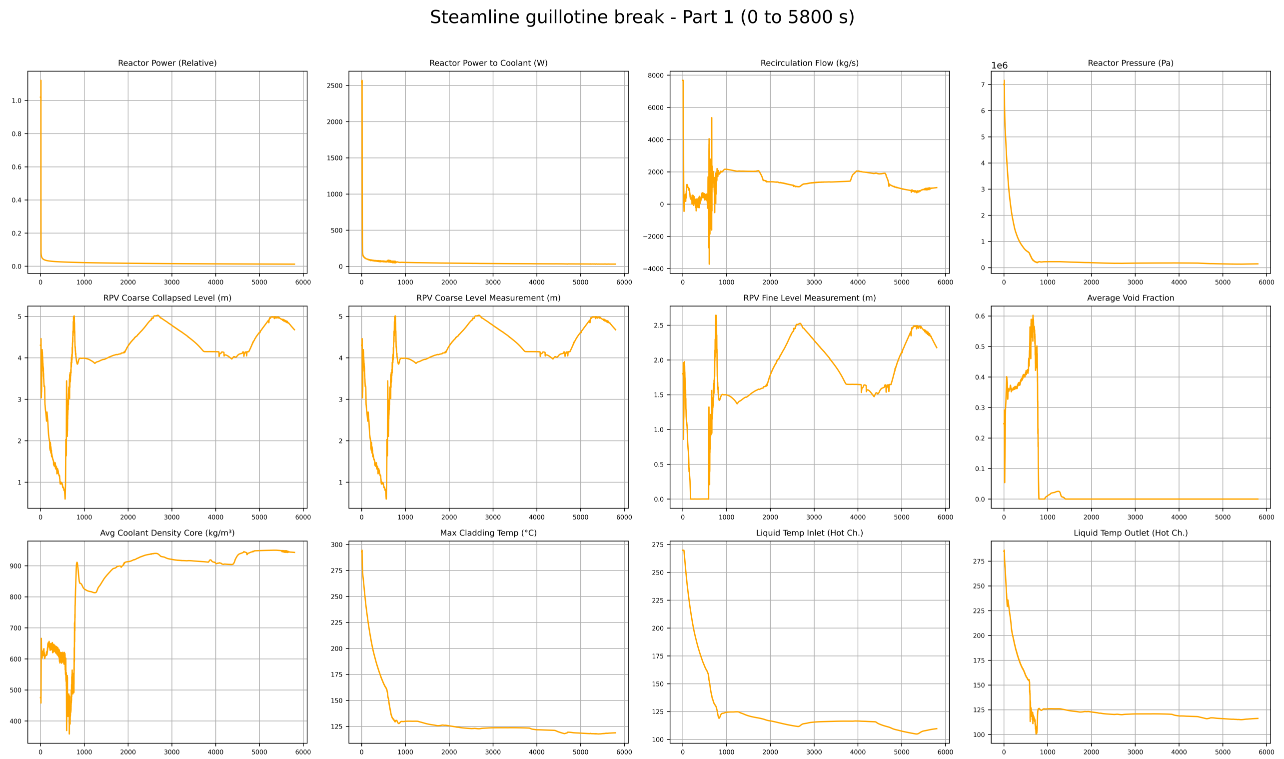


Figure 10: Visual of the 1st set of recorded parameters evolution for the first 5800s of the steamline break accident.

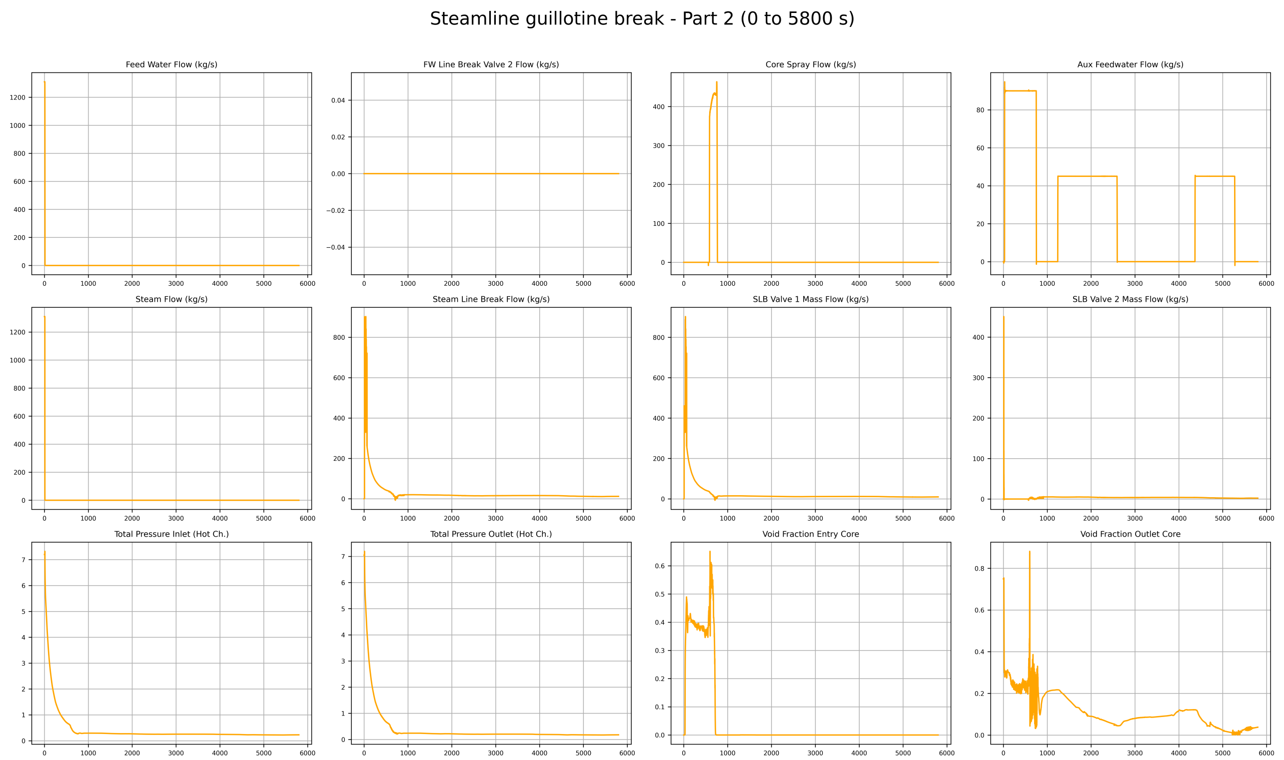


Figure 11: Visual of the 2nd set of recorded parameters evolution for the first 5800s of the steamline break accident.

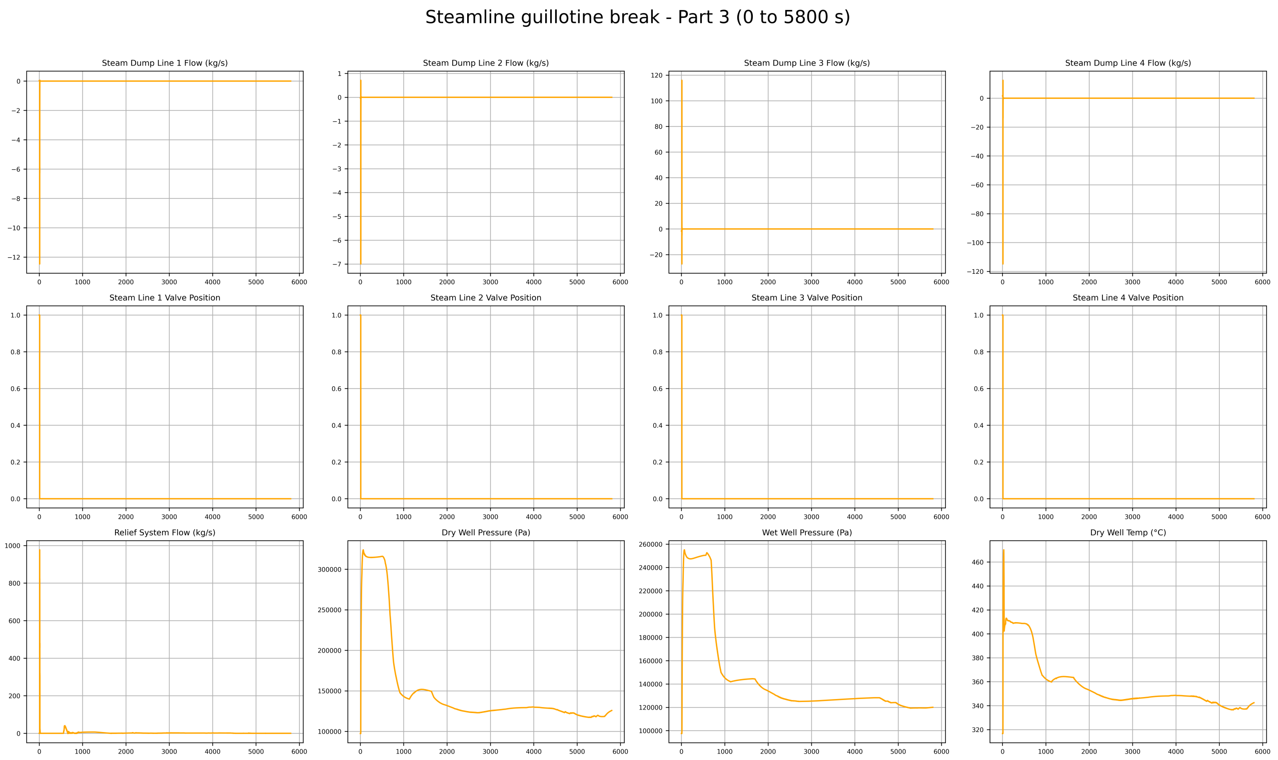


Figure 12: Visual of the 3rd set of recorded parameters evolution for the first 5800s of the steamline break accident.

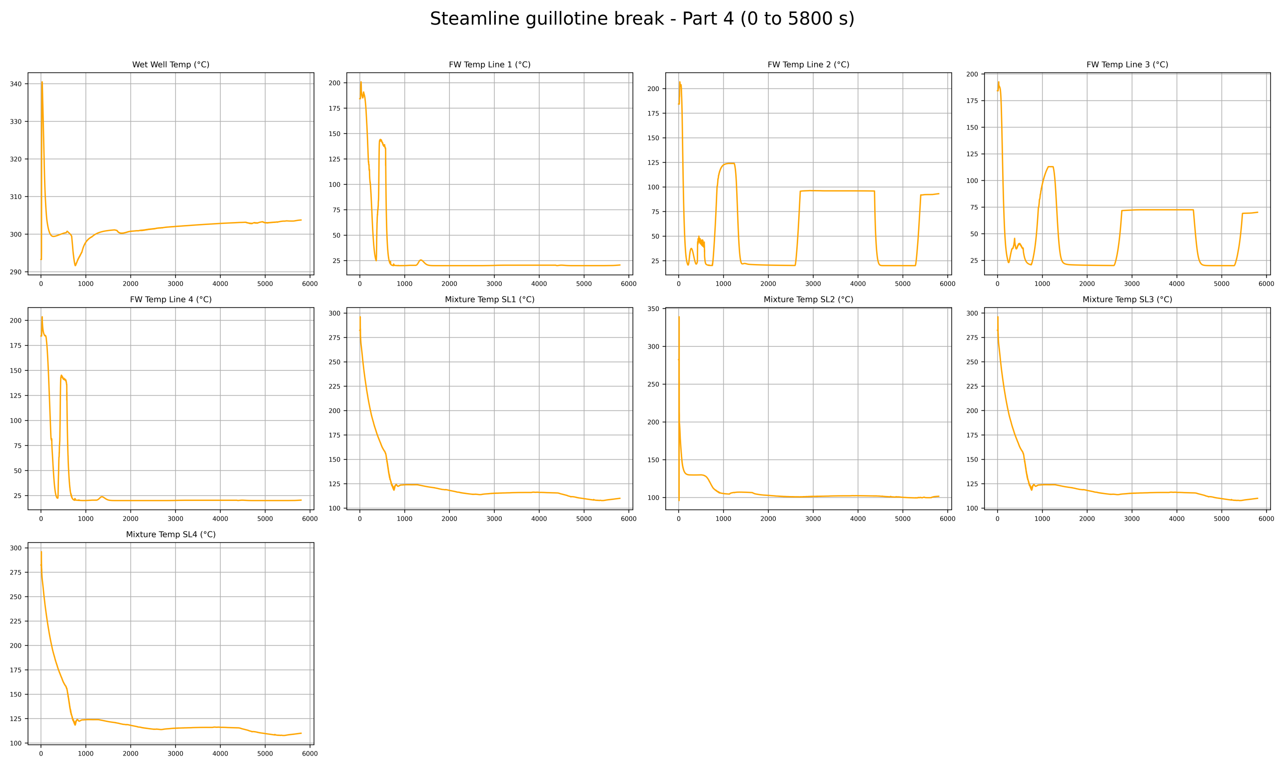


Figure 13: Visual of the 4th set of recorded parameters evolution for the first 5800s of the steamline break accident.

## 3.3 Conclusion

As expected, we had an initial pressure dip but what we did not expect was the quick and short rise in pressure caused by the closure of some of the steam line isolation valves (birefly reducing total steam flow below nominal steam flow).
The system was capable of handling the LOCA and, thanks to the ECCS, the core remained covered for the entire duration of it, preventing a single failure from leading to an aggravated accident.

# 4 Feed water line break

## 4.1 Expectation

Due to the loss of feedwater, there will be a loss in pressure, leading to the SCRAM of the reactor. To enable the ECCS, an opening of the steam relief valves is expected to inject water inside the core to compensate for the water loss and prevent core uncovery and fuel damage. The safety injection systems are expected to activate and restore core cooling.

## 4.2 Results and Discussion

### Short term behaviour

#### 0-60 seconds

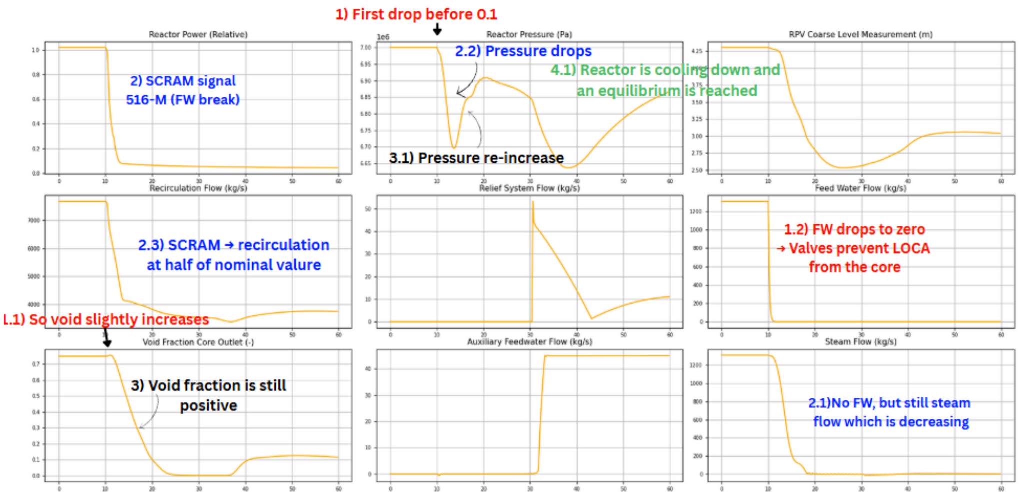


Figure 14: Short term behaviour from 0-60 s

Upon the break of the pipe, valves at the outside of the containment are closed to prevent loss of coolant from the core. SCRAM is initiated, and a significant pressure drop can be seen due to the SCRAM and the lack of fresh feedwater. The recirculation flow is set to half the nominal value, circulating the remaining water that heats up during the process. Due to the lack of feedwater, the water starts to drop, reaching its minimum at 2.5 m at 25 seconds into the transient. 10 seconds after the SCRAM, a turbine trip is initiated, and with a delay of another 10 seconds, the safety relief valves are opened. At the same time, the auxiliary feedwater is injected, flushing the core with fresh, cold water. Due to this combination, the pressure drops, until the water starts to boil again, the pressure rises, and the water level starts to increase again.

### 60-1000 s

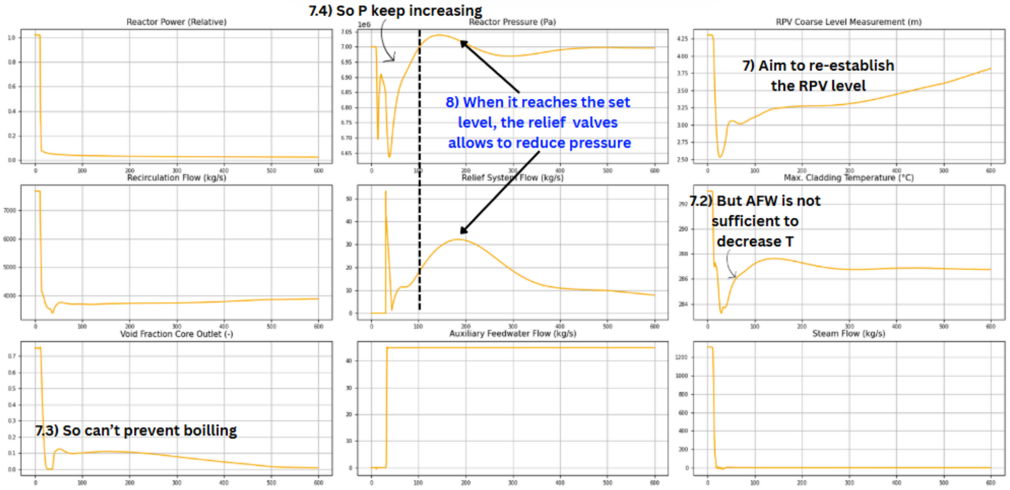


Figure 15: Short term behaviour from 0-600 s

The auxiliary feedwater system continues to supply feedwater for the cooling, increasing the water level slowly with the aim of achieving a stable, high water level. As no steam leaves the vessel, but boiling is occurring, pressure builds up, which gets vented through the relief system to achieve a stable state at 70 bars. The void fraction decreases slowly to 0. The moment a water level of 4.5 m is reached, the auxiliary feedwater is stopped, and due to the high power, boiling starts to occur again, which is consequently compensated by the opening of the relief system to reduce the pressure spike. Due to the high decay heat, the water level drops again.

## Long term behaviour

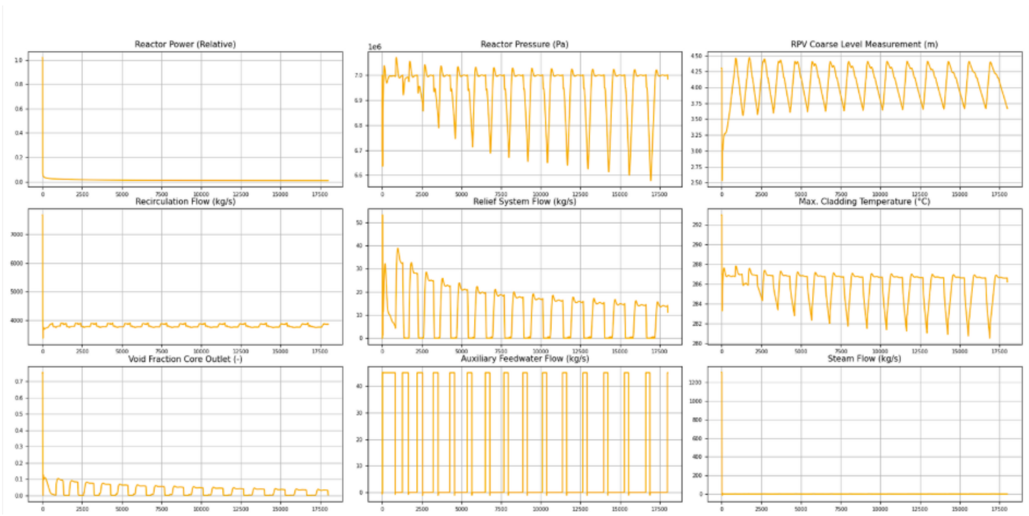


Figure 16: Long term behaviour up to 5 hours

As can be seen in Figure 16 in the long term of 5 hours, there will be an oscillatory behaviour between the water level on top of the core, which activates the auxiliary feedwater if the level drops to 3.7 m, and deactivates when the water level has been restored. The safety relief valves compensate for the steam production when water evaporates again due to the decay heat.

## 4.3 Conclusion

In the end, the water level is still very high, even at the lowest point of the accident, at 2.5 m, which shows the importance of the auxiliary feedwater systems and proves the reactor's resiliency to such an initiating event and, therefore, the importance of defense in depth. The reactor was never close to core uncovery or core damage.

# 5 Feed water pump trip

## 5.1 Objectives and Expectations

The primary objective of this transient simulation is to analyze the dynamic response of a generic nordic Boiling Water Reactor (BWR) to a complete trip of all feed water pumps. Using APROS, we investigate how the reactor behaves under the sudden loss of feed water and how the system's safety mechanisms are activated. More specifically, this transient helps distinguish between a feed water line break and a pumps trip scenario, which is crucial for reactor diagnostics and automated protection response validation.
**Triggering Event:** At time $t = 10$ s, an input signal is sent through the `445_control` diagram that trips all three feed water pumps simultaneously. This input also modifies boundary conditions at the feed water inlet of the pumps, reducing both the enthalpy and pressure of the incoming water.
**Expected Outcomes:**

- The reactor will initiate a SCRAM (reactor shutdown) sequence, with an expected 20-second delay.
- Steam relief valves will open to manage the increasing pressure inside the reactor core.
- The auxiliary feed water (AFW) system will activate in stages to re-establish coolant flow and prevent core exposure or fuel damage.
- Despite the safety systems' intervention, significant degradation of feed water flow is anticipated during the first moments of the transient.

## 5.2 Results and Discussion

### 5.2.1 First 120 Seconds

After running steady state, at $t = 10$ s, the feed water pumps are tripped. As shown in fig:set$_{p}aram_{p}ump_{t}rip_{1}20s,thefeedwaterflowdropsrapidlytozero.Withnofreshwaterenteringthecore,the$
This temperature increase leads to a rise in the average void fraction within the core. Since a higher void fraction reduces neutron moderation, reactor power begins to decrease. Additionally, the increase in coolant temperature causes a reduction in density, which results in an initial drop in recirculation flow.
Simultaneously, the steam flow slightly decreases due to the reduced recirculation. Around $t = 23$ s, the recirculation flow begins to increase again in an attempt to compensate for the decreasing reactor power. This compensatory response is part of the system's feedback mechanisms aimed at stabilizing the thermal output.
At approximately $t = 30$ s, the reactor undergoes a SCRAM, which had been triggered by a 20-second delay following the pump trip. As visible in fig:pump$_{t}rip_{p}ressure_{1}20s,asharppressuredropisobserv$
Following the SCRAM, reactor power rapidly decreases to approximately 5% of its nominal value, driven solely by residual decay heat. The recirculation pumps enter shutdown operation, reducing flow to a minimal level. The void fraction at the outlet also falls to zero after SCRAM since the steam production becomes lower than the ongoing steam removal.
Subsequently, the auxiliary feedwater system is activated in stages. At $t = 40$ s, two out of four auxiliary feedwater lines are activated after the water level drops to 3.6 m. At $t = 60$ s, the remaining two auxiliary lines are triggered when the water level reaches the second threshold of 1.3 m.

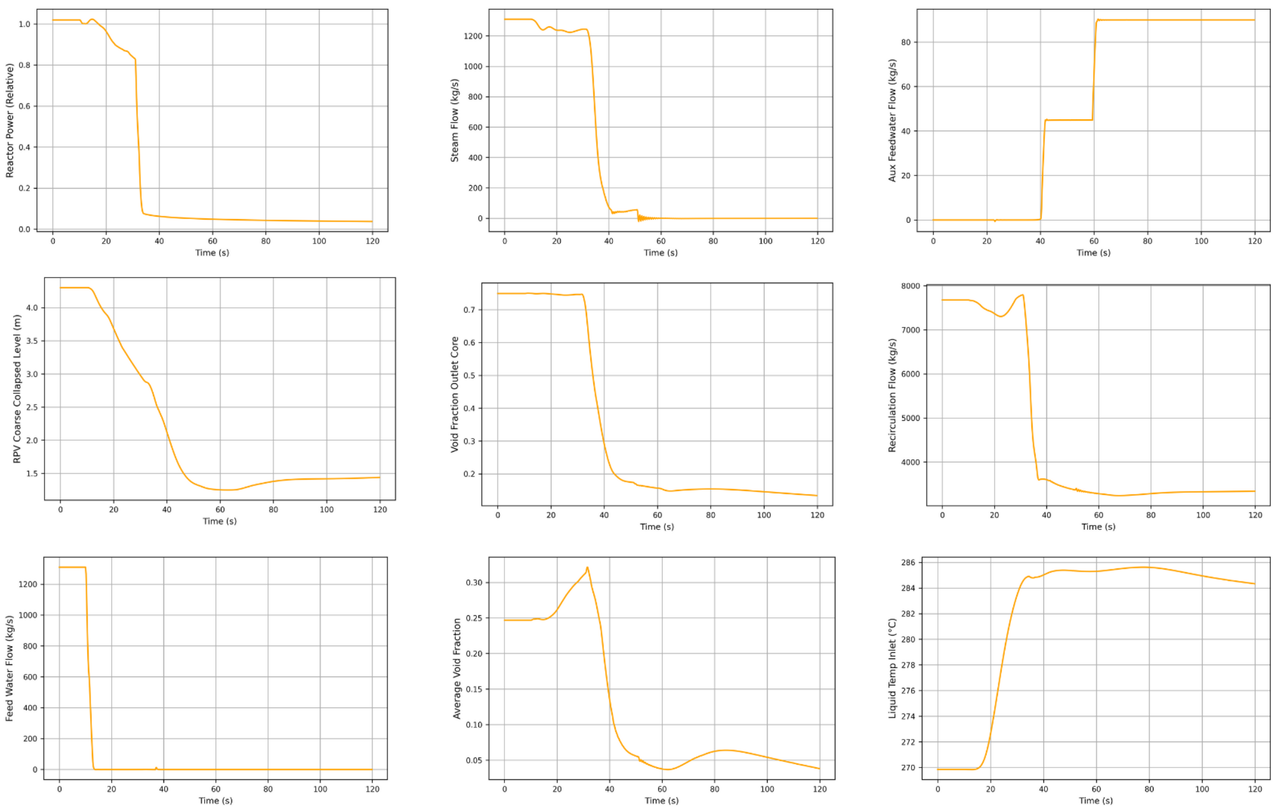


Figure 17: Set of parameters of the reactor during the first 120 seconds of the transient.

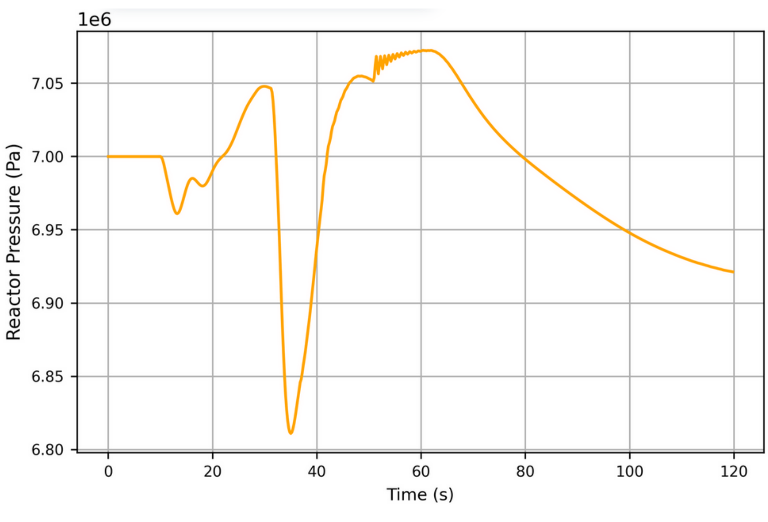


Figure 18: Core pressure evolution during the first 120 seconds following transient.

### 5.2.2 Up to 2500 Seconds

Following the SCRAM at $t = 30$ s, the reactor transitions into shutdown operation. As shown in fig:pump$_t rip_2 500 s_2, thereactorpowercontinuestograduallydecreaseoverthenext2500seconds, correspondin$
A clear correlation can be observed between the core pressure and coolant temperature: both follow the same trend, as shown in fig:pump$_t rip_2 500 s_1. Thisisexpectedsincesystempressureina BWR isclos$
40 s and continuing until roughly $t = 900$ s—causes the overall temperature to decrease, resulting in a corresponding pressure drop.
At $t = 300$ s, the recirculation pumps are set to a minimal rotation speed of 650 RPM, consistent with the reactor's shutdown mode. From shortly after the pump trip, the coolant density

begins to increase as the void fraction decreases. This change reflects the system's thermal stabilization, as voids collapse and liquid water becomes more dominant.
The water level in the reactor initially drops after the pump trip. Once it reaches the first and second threshold levels—$L_1 = 3.6\,\text{m}$ and $L_2 = 1.3\,\text{m}$, respectively—the 2/4 and then 4/4 auxiliary feedwater lines are activated. This leads to a relatively fast reflooding of the core.
Around $t = 2000$ s, the system enters a periodic quasi-steady state characterized by alternating auxiliary feedwater injection and steam relief flow. This regime indicates a balance between heat removal, decay heat generation, and system feedback dynamics.

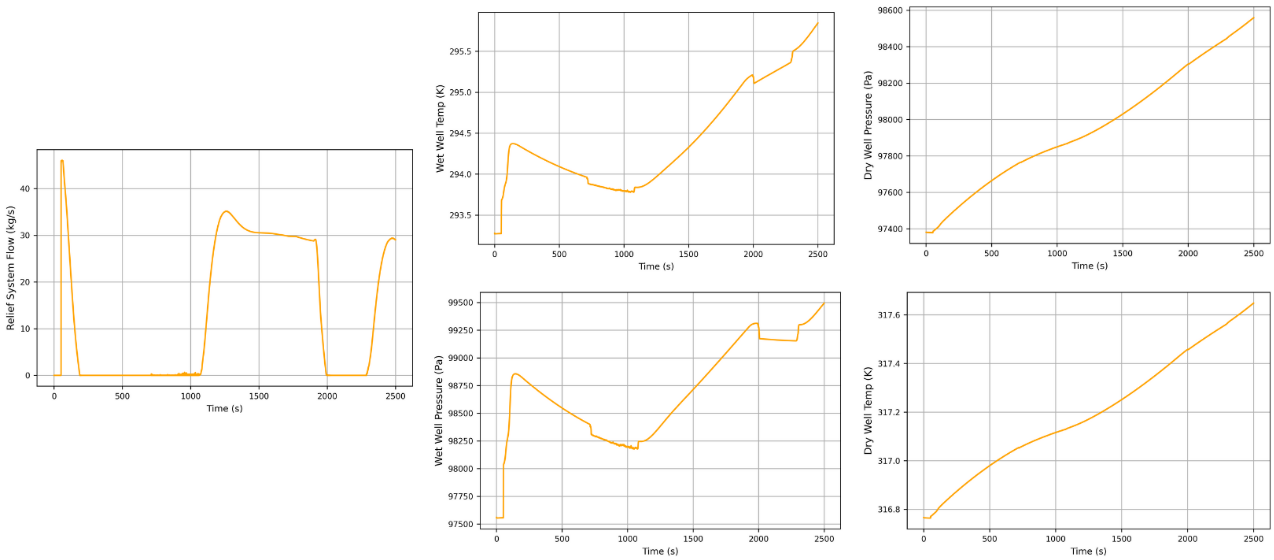


Figure 19: Core pressure, coolant temperature, and void fraction evolution up to 2500 seconds after the feedwater pump trip.

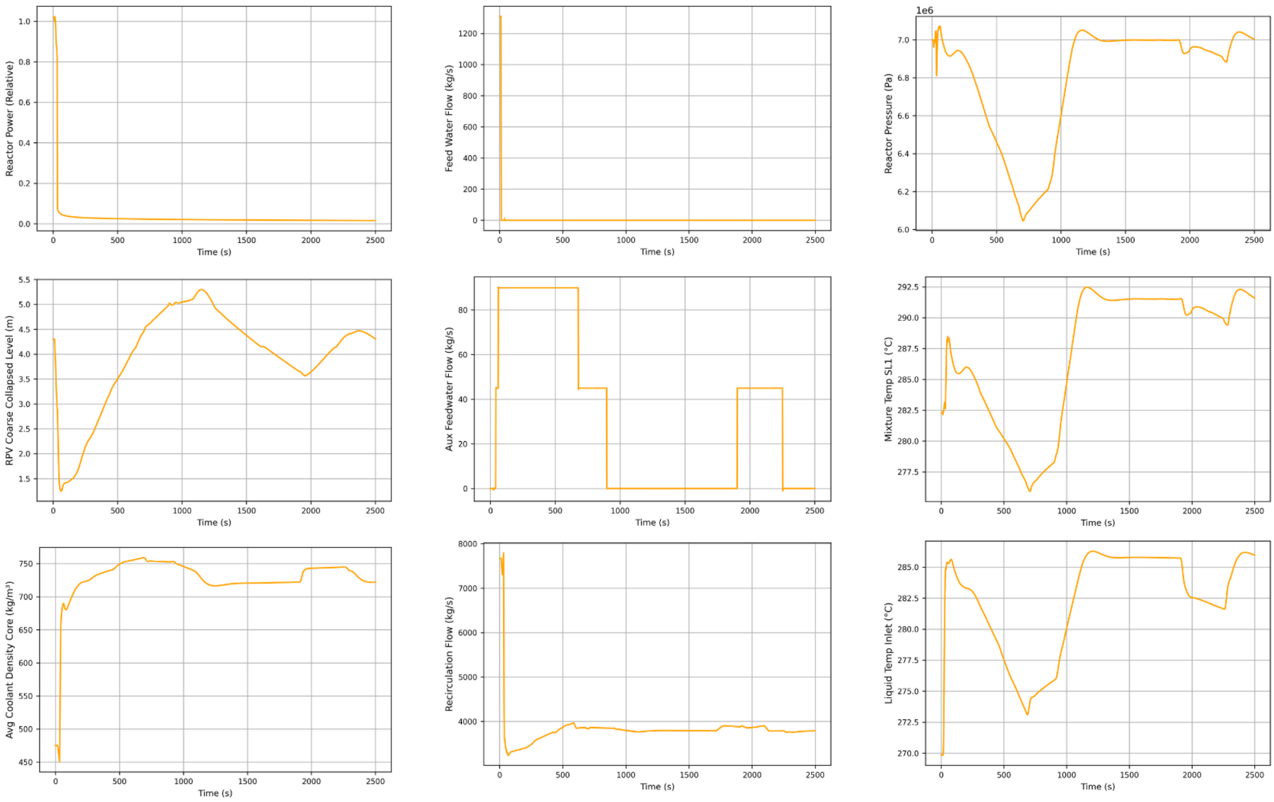


Figure 20: Decay power reduction and recirculation flow behavior during extended shutdown operation.

## 5.3 Conclusion

The simulation results confirm that the Boiling Water Reactor remains safe and stable throughout the feedwater pump trip transient for the simulated time. Despite the sudden and complete loss of feedwater, all primary safety mechanisms were activated as expected. The SCRAM was triggered with a 20-second delay, followed by the staged activation of the auxiliary feedwater (AFW) system and the operation of the steam relief valves to manage pressure buildup.

The analysis aligns well with initial expectations. The reactor power decreased sharply after the SCRAM, and the core was successfully reflooded before reaching critical low water levels. No signs of core uncovering or overheating were observed, and the void fraction returned to safe values as coolant conditions stabilized. These outcomes confirm the system's robustness in handling design basis accidents (DBA) involving single component failures, such as simultaneous feedwater pump trips.

While the transient was largely consistent with predictive models, the simulation highlighted the importance of the timing and interaction of control systems. Specifically, the delay in SCRAM initiation and the staged activation of AFW lines played a critical role in stabilizing core conditions.

Long-term system behavior beyond 2500 seconds shows the onset of a periodic quasi-steady state, characterized by alternating AFW and relief valve activation. A longer simulation, extended to longer time could confirm that the reactor reaches full thermal-hydraulic equilibrium under shutdown conditions without any core damage or safety limit violations.

# 6 Feed Water Enthalpy Decrease

## Remark

For this transient, we encountered issues when attempting to use the gradient module to gradually reduce the feedwater enthalpy. Each time the gradient component was implemented, the simulator—upon starting the simulation—would instantaneously set the inlet temperature to 0 K (the default value in shutdown mode).
This led to unphysical behavior: the sudden injection of near-zero-temperature water caused an immediate and extreme spike in reactivity, triggering an instant SCRAM due to a sharp power surge. Additionally, the transient developed contrary to the intended scenario: instead of observing a decreasing temperature over time, the inlet temperature began to increase (due to the slope direction in the gradient module being inadvertently set "up" toward the target value).
To resolve this, we chose to directly set the inlet enthalpy to the desired value without using the gradient module. As a result, there was no gradual slope in the temperature transition, and the temperature dropped instantly at the beginning of the simulation. This caused a sharp transient instead of a smooth evolution, limiting our ability to study the progressive response of the system to a controlled enthalpy decrease.

## 6.1 Objectives and Expectation

In this scenario, we simulate a transient involving a decrease in the enthalpy of the feed water in a Boiling Water Reactor (BWR). This represents a possible cold shock scenario, such as when water from a non-preheated pipe is injected into the reactor system.

- **Initial condition:** The reactor is operating in a steady state.
- **Action:** Gradually decrease the enthalpy of the feed water. This is achieved by adding a gradient component between `445LTSWI1` and `445LTBC1` in the `445_pump_inlet_control` diagram.
- **Expectation:**
  - The feed water temperature will decrease, but the mass flow rate is expected to remain stable.
  - The reduction in feed water temperature will increase its density, altering the neutron moderation in the core.
  - The core water will cool, resulting in a rise in the water level.
  - The power of the reactor will increase due to the reactivity feedback from the coolant density and temperature change.
  - Ultimately, this power increase may trigger a reactor SCRAM for safety.

In fig:schematic$_f w[H] theschematicofthefeedwatersystemisrepresented.$

### 6.1.1 First 60 Seconds

At $t = 10$ s, the specific enthalpy of the feedwater is reduced, leading to a drop in feedwater temperature to approximately 100°C, as shown in fig:set$_1 p aram_e nthalpy_6 0s.This"coldshock"affectstheinlette$
$hydraulicandneutronicphenomena.$

Due to the increased density of the colder water, the feedwater mass flow increases while the volumetric flow remains controlled. Since the recirculation pumps operate at constant speed (volume control), the increase in density directly translates into higher mass flow (kg/s). This change is evident from $t = 20$ s onward, with recirculation flow progressively increasing up to $t = 30$ s.
At around $t = 28$ s, reactor power begins to rise. This is a result of enhanced moderation caused by the denser coolant entering the core. Initially, the void fraction decreases slightly due to increased condensation associated with the colder water. The water level remains nearly constant up to $t = 40$ s.
Between $t = 30$ and $t = 40$ s, a rapid change in recirculation flow occurs as the system attempts to stabilize the increasing power. This dynamic interaction induces power oscillations, which are characteristic of feedback-modulated control behavior. The void fraction begins to follow the trend of reactor power, leading to oscillations in the average coolant density as well.
At $t = 40$ s, the reactor reaches the SCRAM activation threshold due to the high power level. The shutdown signal is issued, and the transient enters a post-SCRAM phase.
As illustrated in fig:set$_{2p}aram_{e}nthalpy_{6}0s, thereactorpressuredropsbyapproximately3.2barimmediatelyf$
$thesteamcontinuestoexitthesystemfasterthanitisbeinggenerated, especiallyasthepowerdropssharply.Alt$
Following the SCRAM:

- The recirculation pumps transition to minimum-speed shutdown mode.
- Feedwater injection is stopped, and the reactor enters shutdown operation.
- A drop in water level is observed, resulting from increased coolant density and reduced incoming mass flow.
- Inlet temperature shows a change in slope, reflecting the cessation of feedwater flow.
- Steam flow and void fraction both approach zero.
- Coolant density increases further as the void content diminishes.

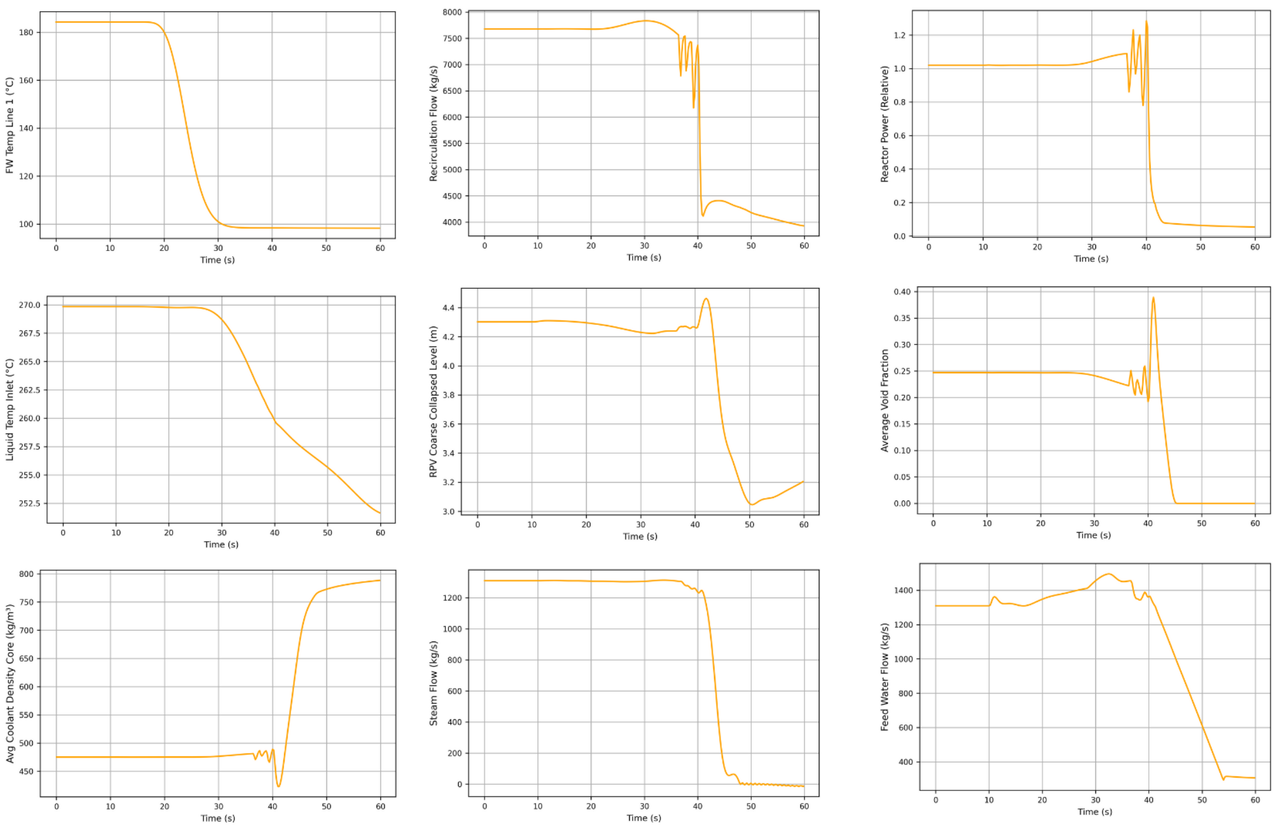


Figure 22: Feedwater temperature, recirculation flow, and reactor power during the first 60 seconds following the enthalpy decrease.

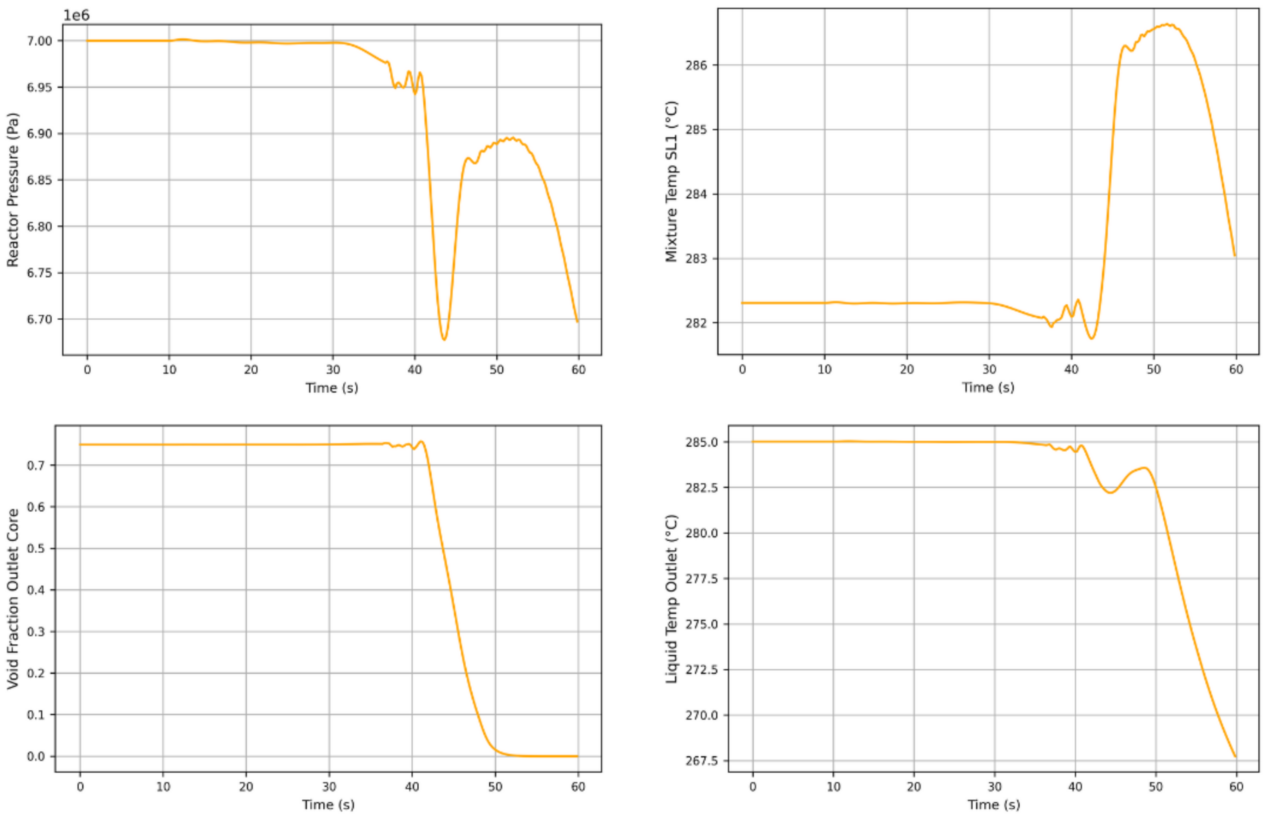


Figure 23: Pressure and void fraction behavior after SCRAM during the feedwater enthalpy transient.

### 6.1.2 First 120 Seconds

Following the initial 60 seconds of the feedwater enthalpy decrease transient, the reactor system continues to respond to the SCRAM and the thermal imbalance caused by the introduction of cold feedwater.
As shown in fig:set$_{1p}aram_enthalpy_120s, reactorpowerdecreasesgraduallytowardszeroduetoresidualdecayh$
At approximately $t = 60$ s, two out of four auxiliary feedwater (AFW) lines are activated. This action is triggered by the pressure relief system, which responds to elevated core pressure. The water level within the reactor vessel begins to rise due to the combined effect of both auxiliary and primary feedwater flows. A noticeable change in slope appears in the water level curve as the feedwater system transitions from low-power to full-power mode once the L2 threshold (3.6 m) is reached.
Some oscillations and noise are observed in the AFW flow rate during this transition, indicating control fluctuations as the system stabilizes its response to the sharp power drop and water level dynamics.

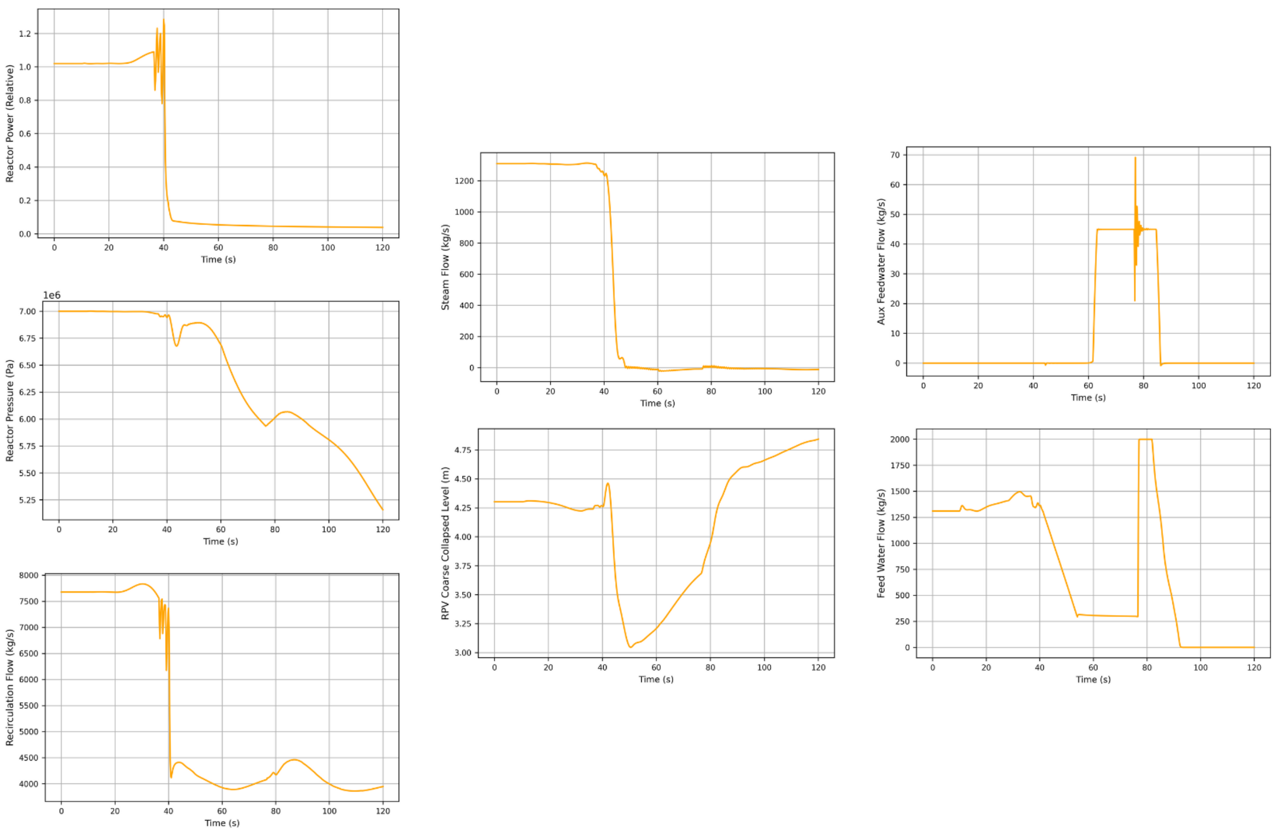


Figure 24: Evolution of power, pressure, recirculation flow, and water level during the first 120 seconds following the feedwater enthalpy decrease.

As depicted in fig:set$_{2p}aram_enthalpy_120s, thepressurereliefsystemactivatesatt = 60$ s, triggered by the pressure rise within the core region. This results in steam being vented into the wet well, which is connected to the dry well, effectively managing the internal pressure. This steam release causes localized increases in both temperature and pressure downstream of the relief valves.

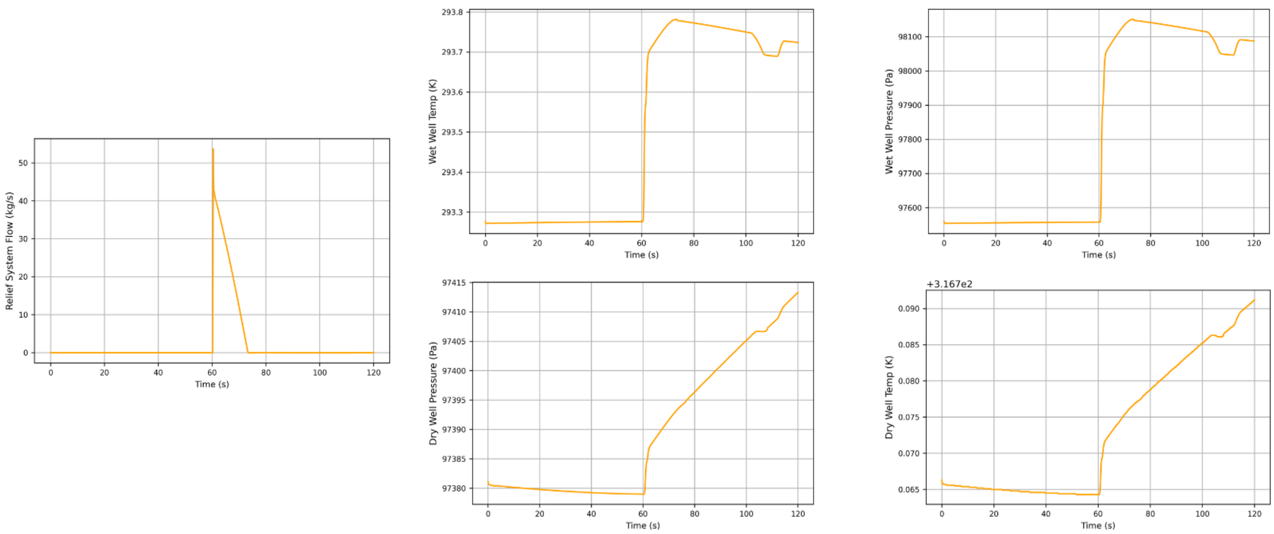


Figure 25: Activation of the relief system at 60 seconds and corresponding effects on pressure and temperature due to steam venting into the wet well.

### 6.1.3 First 2500 Seconds

After the initial transient behavior observed in the first two minutes, the system transitions into a long-term stabilization phase, that we extended to $t = 2500$ s during our simulation.
As shown in fig:set$_{1p}aram_enthalpy_2500s, reactorpowerremainsatzeroandsteamflowstaysnullthroughoutt$
During the early part of this interval, the water level in the reactor vessel continues to rise gradually due to the combined action of residual feedwater and auxiliary feedwater (AFW) injections. After approximately $t = 1500$ s, a periodic oscillation pattern begins. This behavior results from the automatic activation and deactivation of the AFW system based on threshold crossings: AFW turns on when the water level drops below 3.6 m and off when it reaches 4.5 m. These control cycles create a repeating water level sawtooth pattern, reflecting the balance between injection and steam venting dynamics.
The system pressure, having previously dropped following the SCRAM, begins to recover and reaches its original steady-state value by approximately $t = 600$ s. This pressure recovery is aided by the reduced void content and stabilized temperature profile.
The relief system also enters an oscillatory operating mode. It is triggered intermittently whenever the void fraction exceeds its operational threshold, thus preventing overpressurization of the core region by discharging steam into the wet well.
Throughout this phase, the recirculation pumps remain in shutdown mode, operating at a minimal speed of approximately 650 RPM. The flow rate variations during this period are primarily driven by changes in coolant density rather than pump control, reflecting the ongoing thermal relaxation of the system.
As illustrated in fig:set$_{2p}aram_enthalpy_inlet_temp_2500s, theinlettemperatureatthebottomofthefuelassembl$
600 s—coinciding with the pressure stabilization—indicating a new thermal equilibrium at the core inlet.

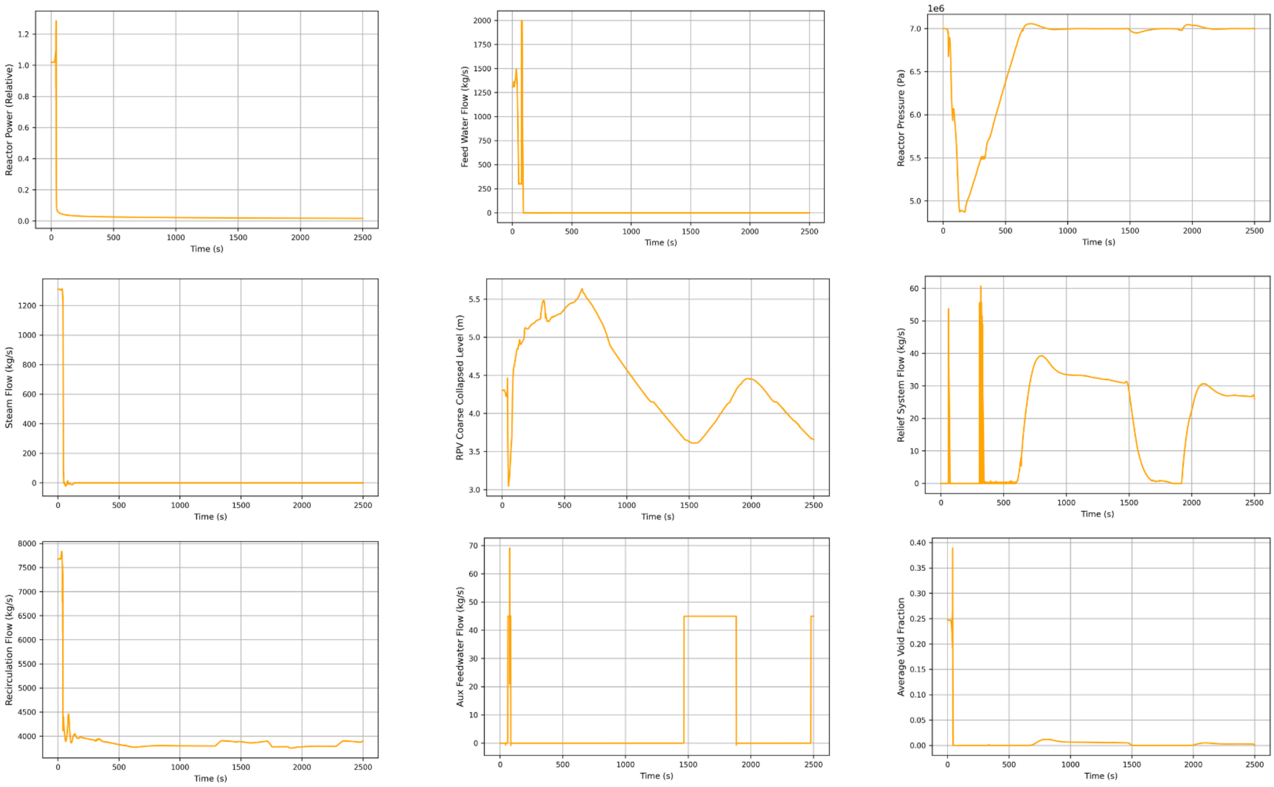


Figure 26: Evolution of reactor power, steam flow, water level, and relief valve activity during the first 2500 seconds of the enthalpy decrease transient.

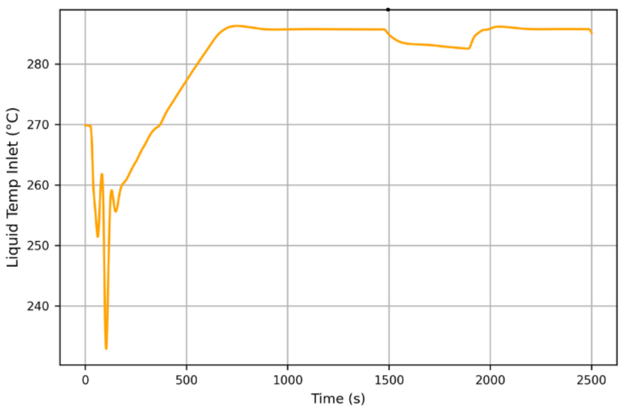


Figure 27: Stabilization of inlet temperature at the bottom of the fuel assemblies after 600 seconds, aligning with core pressure recovery.

## 6.2 Conclusion

The feedwater enthalpy decrease transient, representing a cold shock event in a BWR, revealed a dynamic interplay between thermal-hydraulic and neutronic feedback mechanisms. The gradual reduction in feedwater enthalpy successfully decreased the inlet temperature to approximately 100°C, resulting in increased feedwater density and a corresponding rise in mass flow rate through the core.

This denser coolant improved neutron moderation, leading to a noticeable increase in reactor power. Around $t = 40$ s, the power reached the SCRAM threshold, prompting an automatic shutdown of the reactor. The SCRAM was followed by a significant pressure drop (approxi-

mately 3.2 bar), caused by steam removal exceeding steam production, and a near-complete loss of void content in the core.
Post-SCRAM, the reactor entered shutdown operation. Recirculation pumps reduced to minimal speed, steam production ceased, and power dropped to decay heat levels. The auxiliary feedwater system was activated in stages as water levels decreased, with notable flow oscillations starting around $t = 1500$ s. These oscillations were driven by the AFW system responding to water level thresholds between 3.6 m and 4.5 m.
The relief system intermittently discharged steam to control pressure peaks resulting from void fraction fluctuations. By approximately $t = 600$ s, both pressure and core inlet temperature had stabilized, indicating that the system had reached a new equilibrium state.
Overall, the simulation confirmed the effectiveness of the reactor's safety and control systems in managing a cold feedwater transient. The SCRAM was successfully triggered by the power excursion, and auxiliary systems maintained safe operating conditions thereafter. This transient underlines the importance of monitoring enthalpy and density variations in feedwater, as they can induce significant reactivity effects within the reactor core.

# 7 Reactor power decrease to 90%

## 7.1 Expectation

It is expected that due to the change in the recirculation pump speed, less water is getting pushed through the core, lowering the power. This affects steam production, the feedwater flow, pressure, and void fraction.

## 7.2 Results and Discussion

Due to the reduction in the recirculation pump speed, the reactor power follows immediately, increasing the void fraction and lowering the coolant density. The steam flow accordingly decreases to 90 % of its nominal power. The pressure and coarse water level are nearly constant during this transient with minimal fluctuations. To explain the oscillations in the feedwater, the first idea was to look at the pressure drop by taking the pressure at the inlet of a hot channel and subtracting the outlet pressure of the channel, which proved to be only a minor change. The liquid temperatures at the channel inlet were nearly constant over the transient. It is thought that the oscillations are caused by the input of PIP controllers for the feedwater control, which compare on the one hand the water level with a setpoint, as well as the steam flow with the measured water flow. The response of the controllers to keep the water level constant and adjust to the reduced steam flow will be oscillatory and stabilizes at the lower feedwater flow level after about 60 seconds.
It can be observed that neither the relief nor the auxiliary feedwater system was to be activated. The transient stabilizes at the lower power level and stable parameters in the long run.

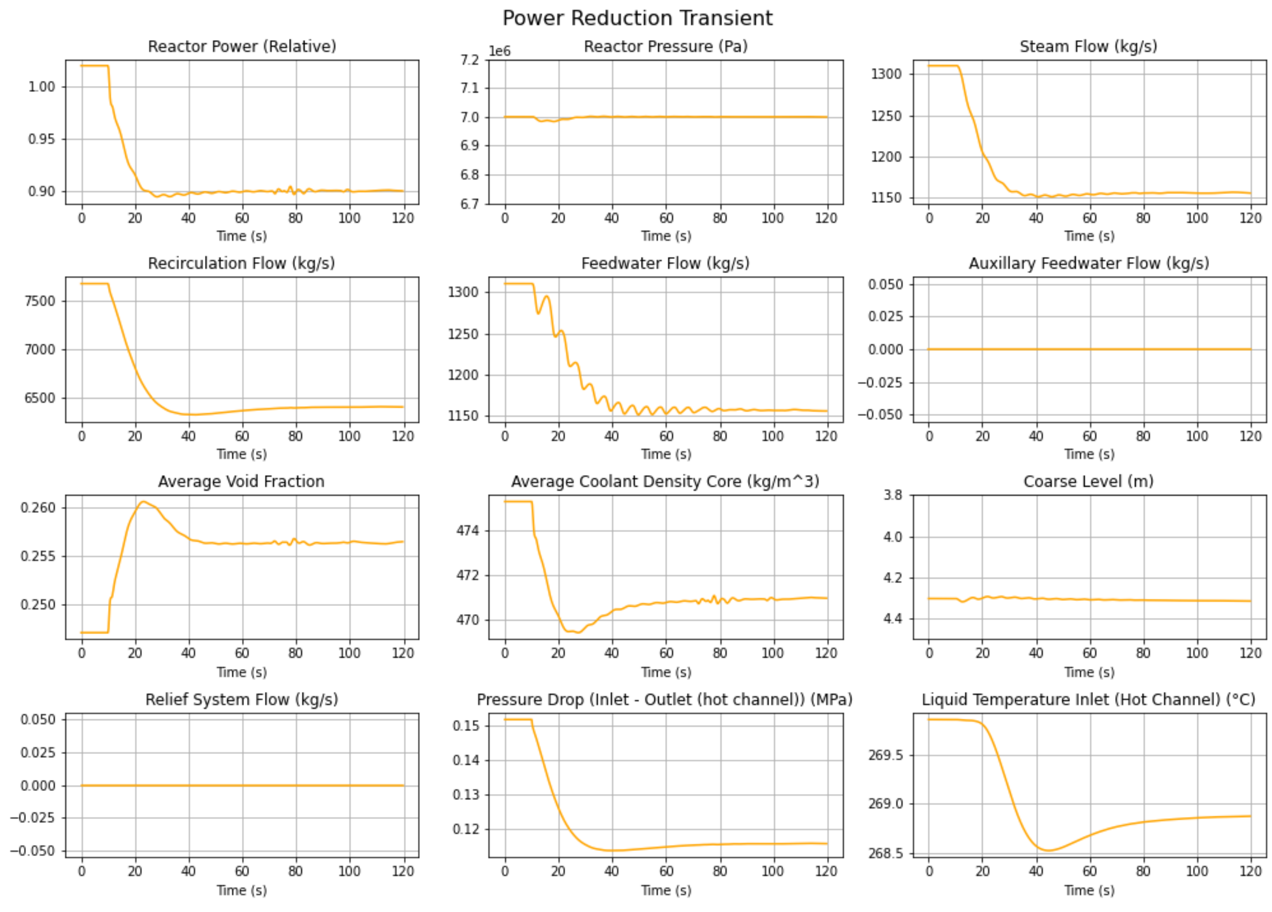


Figure 28: Caption

## 7.3 Conclusion

During this transient, neither the relief nor the auxiliary feedwater system was activated, reflecting that power changes are made by adjusting the recirculation flow rather than with control rods. This transient will lead to a stable state of the reactor, not causing a threat to safety.

# 8 Project

We chose to aggravate the steam line guillotine break by preventing core spraying (with water from the wetwell, as part of the auxiliary feedwater in the model goes through the core spraying lines).

## 8.1 Expectation

We don't exactly know what to expect and are looking forward to the results of the simulation. The project is an aggravated case of the second transient (200% steamline guillotine break) where the water level dropped from the nominal $\tilde{5}$m to 0.7m. At 0.7m, relief and core spraying activate. This causes the water level to increase very rapidly and prevents the core from being uncovered.
Now, without the core spraying, only the auxiliary feedwater refills the vessel and we think that the level will fall below 0.7m and maybe below 0, uncovering the core.
In any case, there is a moment when the time-depending steam mass flow generated by the decay heat becomes lower than the time-independent auxiliary feedwater mass flow and the vessel should start to refill, making the water level rise and the core covered. However, the core could get damaged in the meantime (with the issues that would arise from it).

## 8.2 Results and Discussion

To prevent the core spraying system from activating, the model's control logic was modified. The difference between the original control logic and the modified control logic can be seen on figures 29 and 30 respectively.

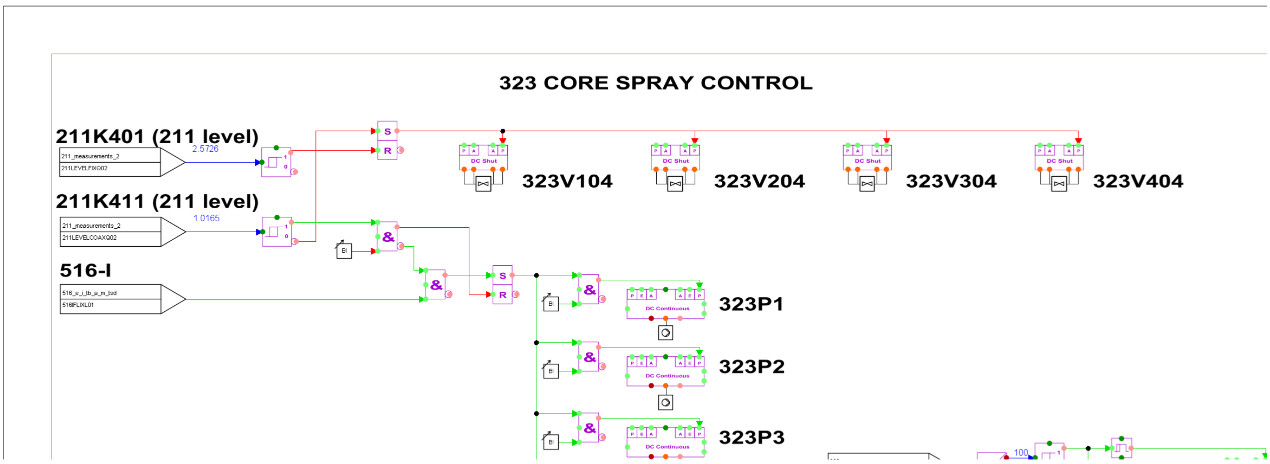


Figure 29: Core spraying control logic before modification

The control operates on a flip-flop logic. The isolation valves are closed with a FALSE signal until the coarse water level drops to 0.7m and SETs the signal to TRUE, opening the valves. They will remain open until the fine water level reaches a set point which would RESET the signal to FALSE and close the valves.

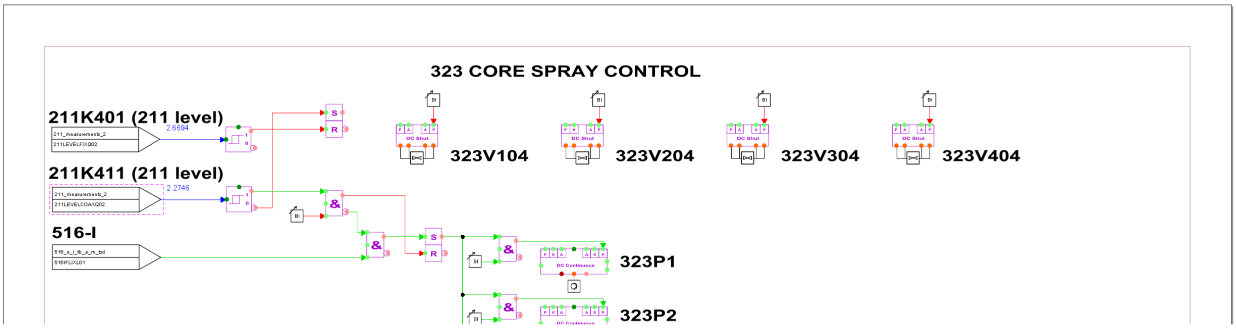


Figure 30: Core spraying control logic after modification

The modification forces a FALSE signal in the valves.
**0 to 5800s summary**

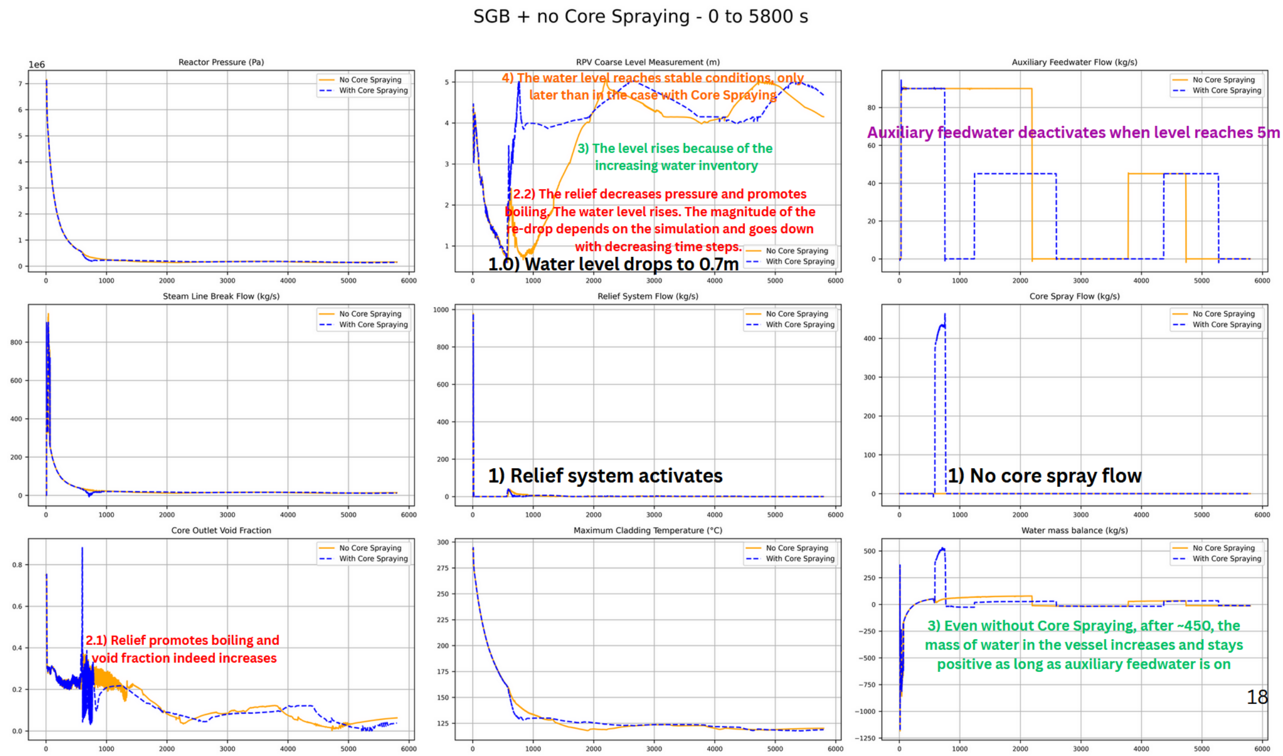


Figure 31: Visual of the key parameters evolution for the first 5800 seconds of the steamline break + no core spraying accident.

This time, when the core water level drops to 0.7m, the relief system also activates but not the core spray, whose flow stays 0.
What happens is that the steam relief cause a drop in pressure, which promotes boiling and makes the coarse water level rise.

The water level eventually drops again until it slowly increases, as the net water mass flow is positive.
Without the core spraying, the water level reaches 5m at 2200s, instead of at 750s with core spraying. The rest of the transient is similar to the 2nd transient, with the water level oscillating between 2 setpoints.
However, what happens after the water level drops to 0.7m is a bit weird and not perfectly understood. Also, depending on the simulation, the water level does not always redrop and sometimes just slowly rises to 5m (with smaller timesteps than in the shown simulation).
This led us to wonder if 0.7m is an Apros limit that messes with our simulation or if the simulation is more or less correct.
To test this, we ran another simulation where auxiliary feedwater flow is removed at 560s, on top of the core spraying removal (essentially, a Loss Of Coolant Accident + Station Blackout). This way, no water can refill the vessel and we will see if the core can become uncovered in Apros.
The control logic for the auxiliary feedwater valves was changed from 32 to 33.

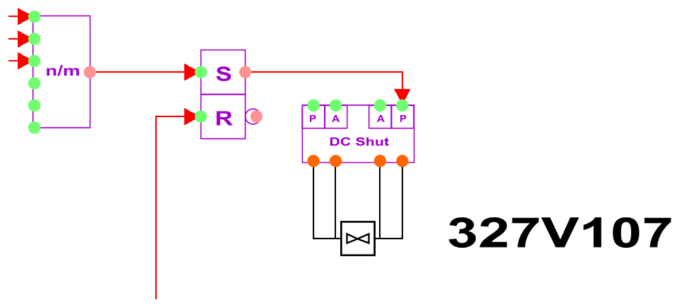


Figure 32: Auxiliary feedwater control logic before modification

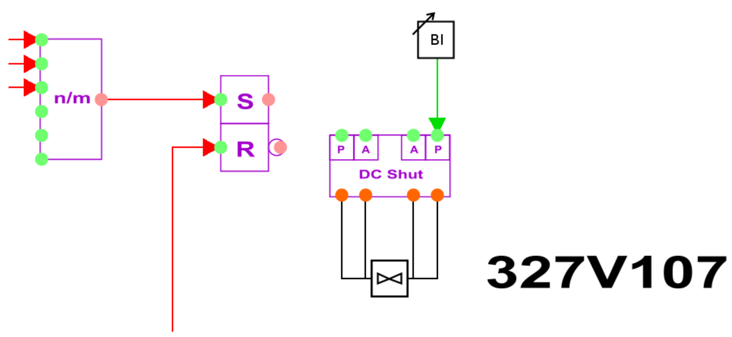


Figure 33: Auxiliary feedwater control logic a after odification

**0 to 3600s summary**
As in the previous simulation, the water level rises after having dropped to 0.7m.
However, this time the auxiliary feedwater is removed (net water flow becomes and stays negative) at 560s and the water level re-drop just continues until reaching a floor at 0.2m, definitely an Apros limit.
After, the void fraction begins to grow until reaching unity, which means that the core is completely uncovered and the cladding temperature eventually skyrockets (meltdown).
**All records**

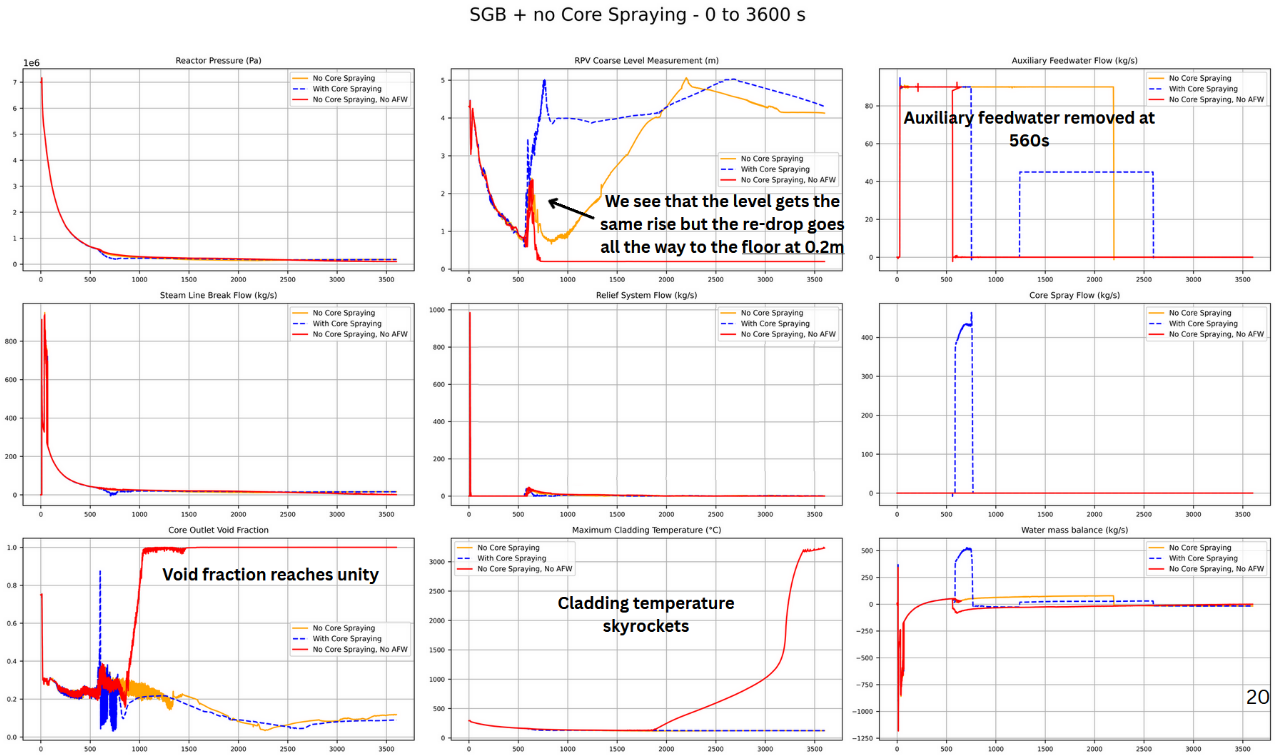


Figure 34: Visual of the key parameters evolution for the first 3600 seconds of the steamline break + no core Spraying + no auxiliary feedwater accident.

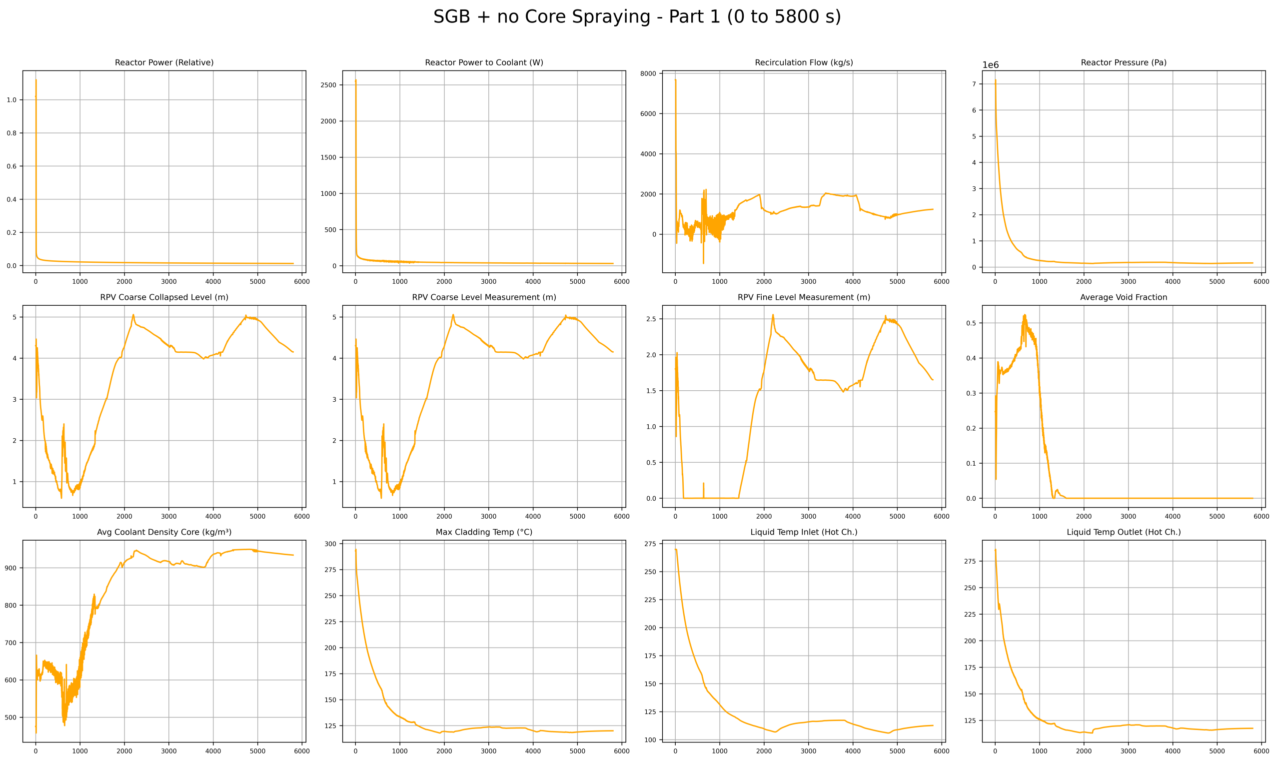


Figure 35: Visual of the of the 1st set of recorded parameters evolution for the first 5800s of the steamline break + no core spraying accident.

### 8.2.1 Conclusion

Apros does not not have a coarse water level floor at 0.7m but at 0.2m.
It seems that auxiliary feedwater alone, without core spraying, is sufficient to completely flood

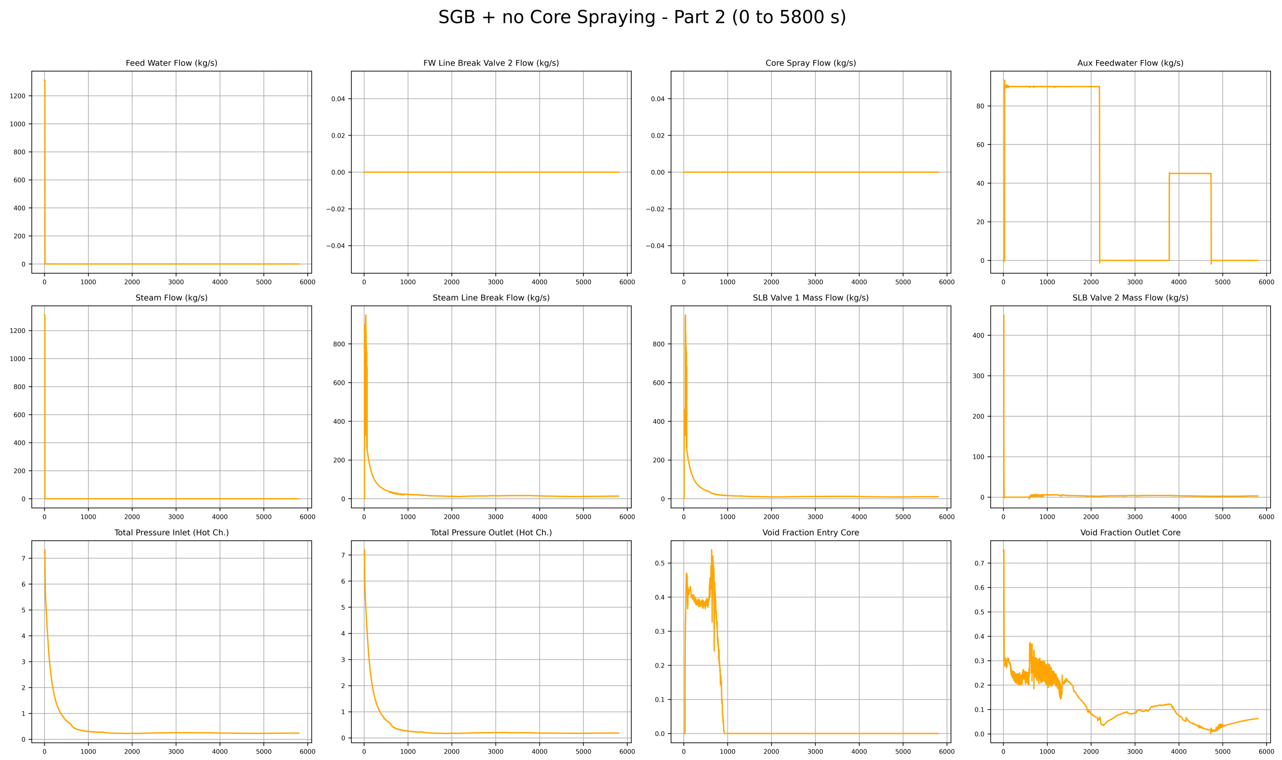


Figure 36: Visual of the of the 2nd set of recorded parameters evolution for the first 5800s of the steamline break + no core spraying accident.

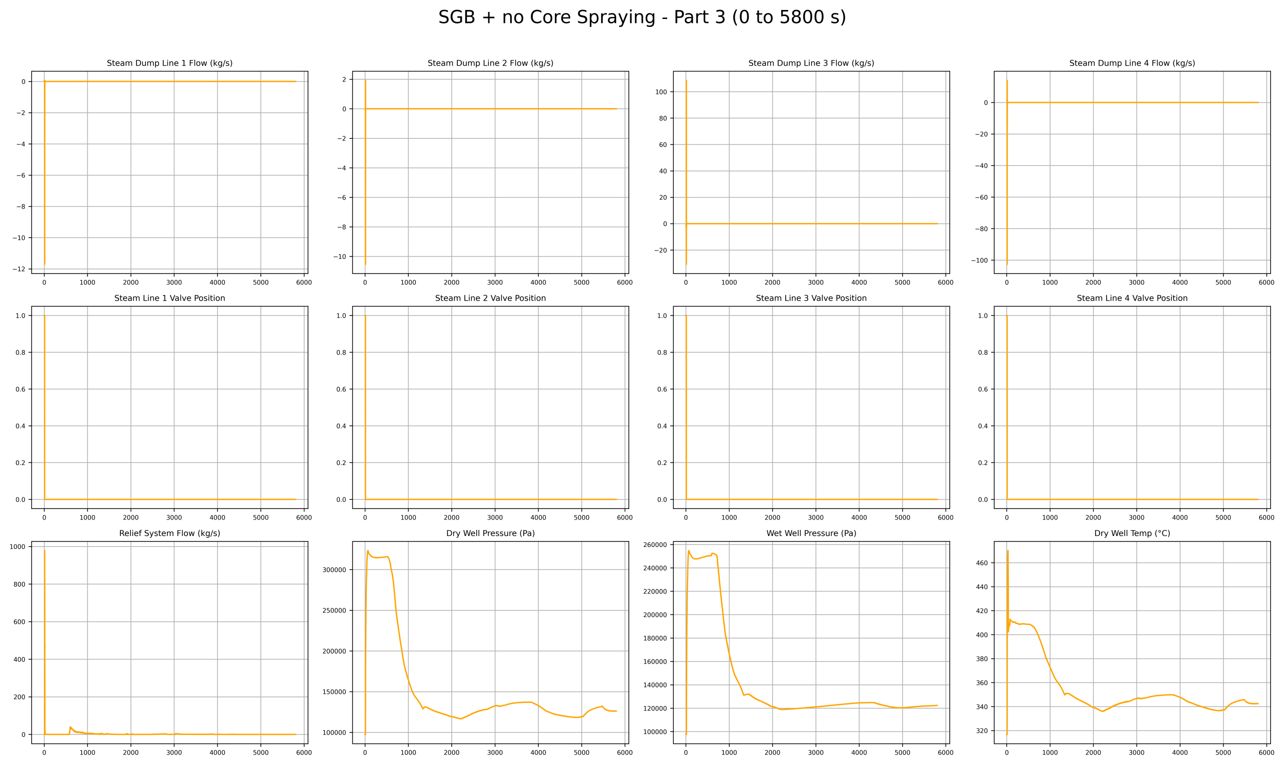


Figure 37: Visual of the of the 3rd set of recorded parameters evolution for the first 5800s of the steamline break + no core spraying accident.

the core, even if it does so slower than with the help of the core spraying.
Delving deeper in Apros should be done to better understand why the simulations vary so much after the coarse water level reaches 0.7m for the first time.

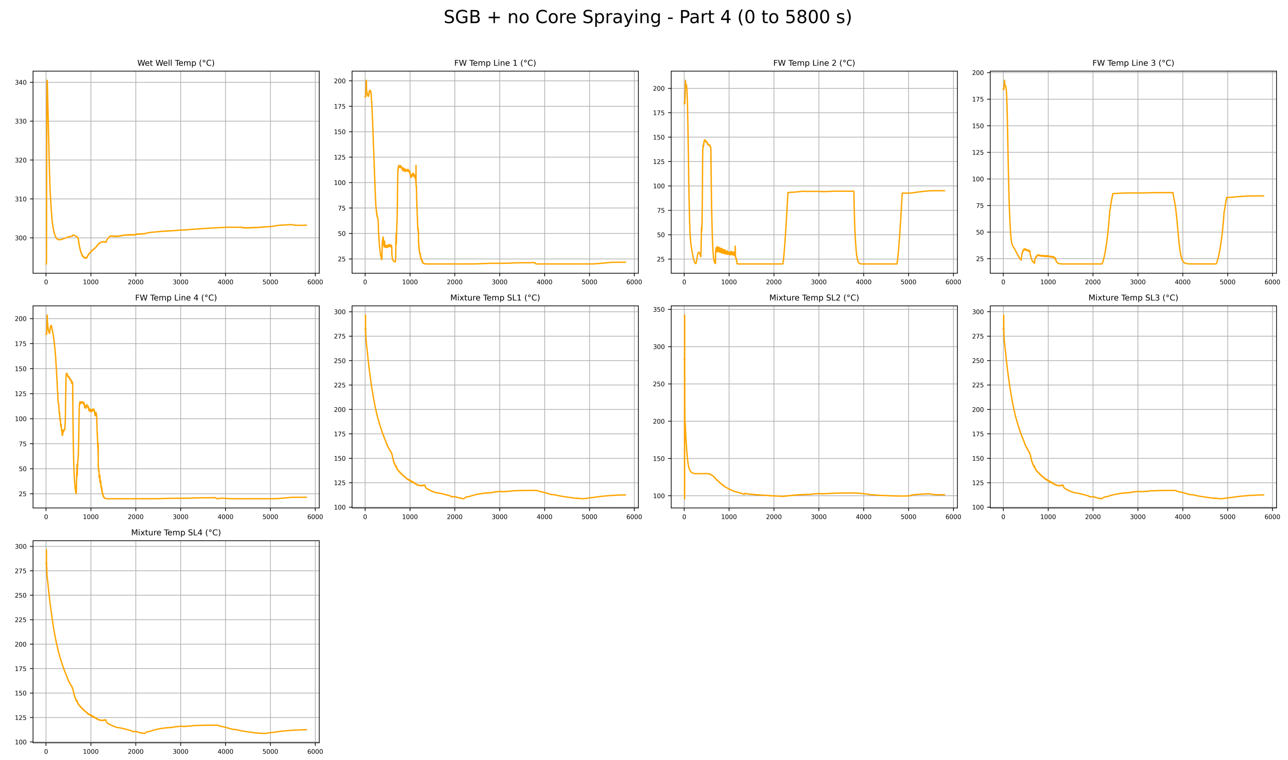


Figure 38: Visual of the of the 4th set of recorded parameters evolution for the first 5800s of the steamline break + no core spraying accident.

Preventing core spraying **and** auxiliary feed water leads to core uncovery and meltdown.