SH2705 Compact Reactor Simulator- Exercises in Reactor Kinetics and Dynamics

Project Report: Interpretations of Reactor Behavior

Aurora Jahan, Faisal Ahmed Moshiur

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Chapter 1

Introduction

This scientific report aims to investigate and predict the behavior of a generic Nordic Boiling Water Reactor (BWR) under two specific transient scenarios. The reactor model used for simulation consists of a core length of 3.7 meters and operates at a nominal power of 2500 MW. The reactor dynamics include six delayed neutron feedbacks, and decay heat is simulated according to the ANSI/ANS-5.1-1979 standard, correlated to the nominal power.

Reactor model and parameters

The simulation model for the reactor is based on the diffusion approximation with two energy groups (fast and thermal neutrons) to approximate the neutron flux. The reactor dynamics are time-dependent, and decay heat is accounted for according to the ANSI/ANS-5.1-1979 standard, correlated with the nominal power.

The reactor simulation model incorporates various scramming mechanisms directly coupled with accident-initiating events and deviations of key process parameters. The model includes important plant parameters and SCRAM limits, some of which are as follows:

Table 1.1: Plant parameters

Parameter	Value
Power	2500 MW_{th}
Pressure in steam dome	7.0 MPa
Recirculation flow rate	7800 kg/s
Feedwater flow rate	1350 kg/s
Water level above the core	4.3 m
Feedwater temperature	185 °C
Safety Release Valves flow rate	1200 kg/s

Table 1.2: SCRAM Limits and Properties

Property	SCRAM limit
High nominal power	$> 106\%$
Low coolant level	< 3.1 m
High coolant level	> 5.0 m
Steam Pressure	> 7.4 MPa
Wetwell liquid temperature	> 35 °C

Chapter 2

Case 1- manual activation of the reactor SCRAM

2.1 Description

In this case, we consider a scenario where operators manually scram the reactor during regular operation at nominal power. The manual SCRAM initiates a rapid shutdown of the nuclear reactions, leading to a series of changes in various system parameters within the reactor.

2.2 Results

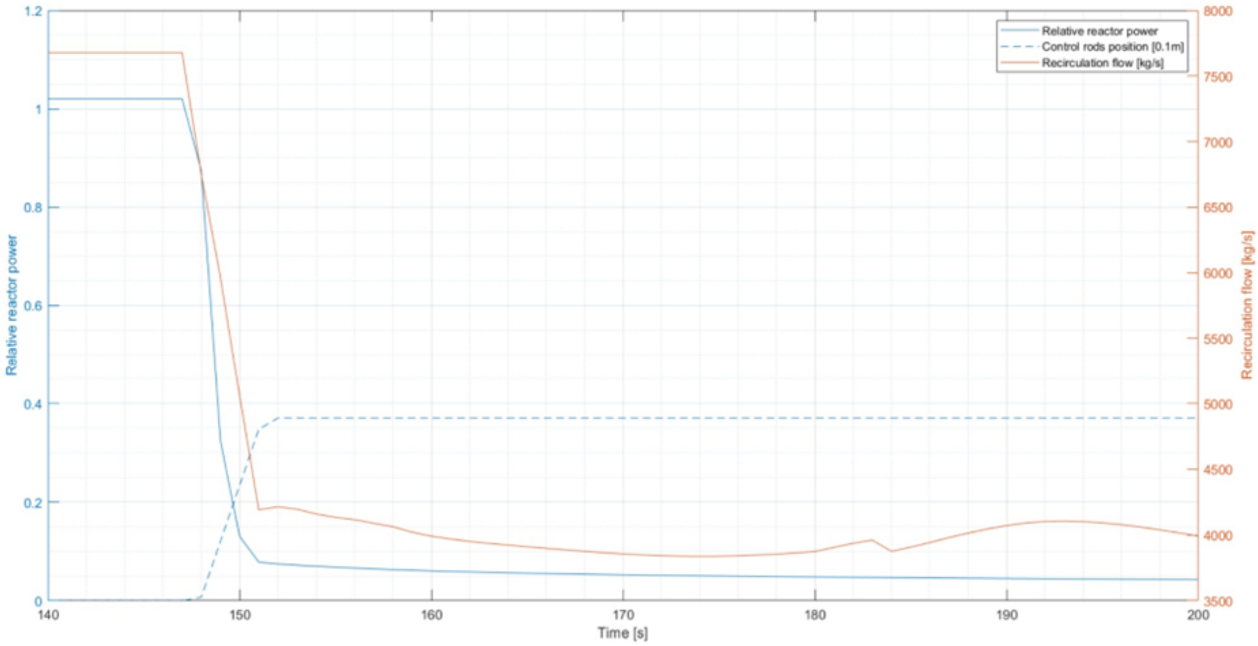

Figure 2.1: Reactor relative power, control rod position, and recirculation flow

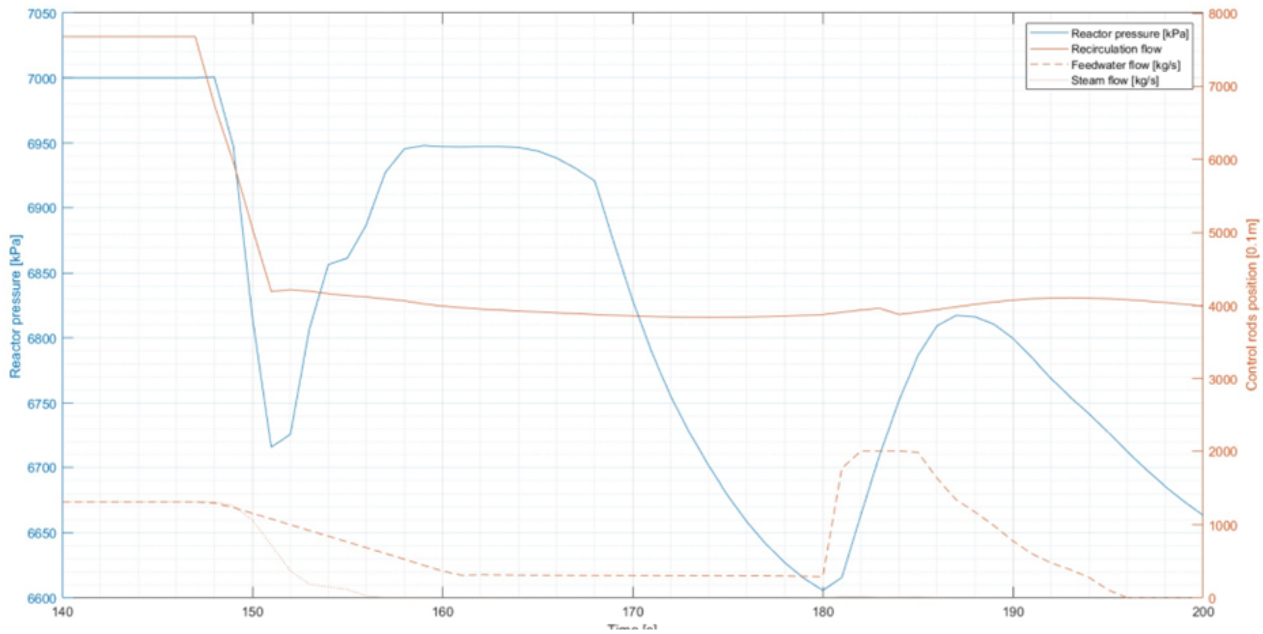

Figure 2.2: Pressure, recirculation flow, feedwater flow, and steam flow

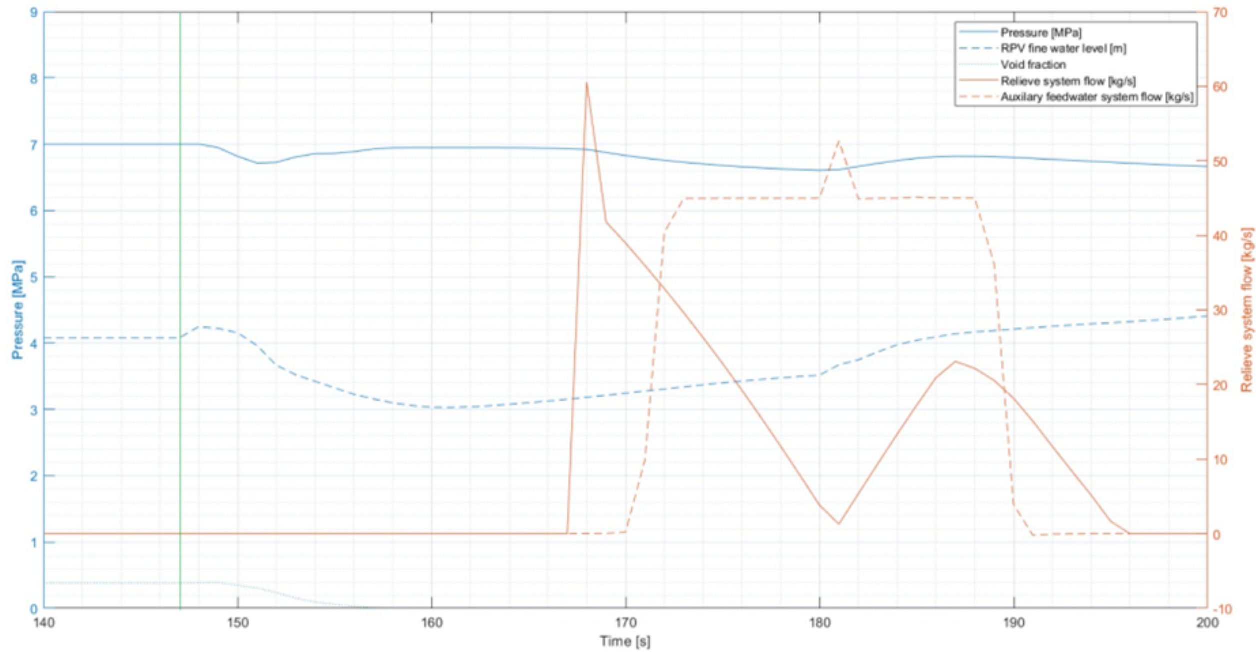

Figure 2.3: Pressure, water level, void fraction, relief system flow, and auxiliary feedwater system flow

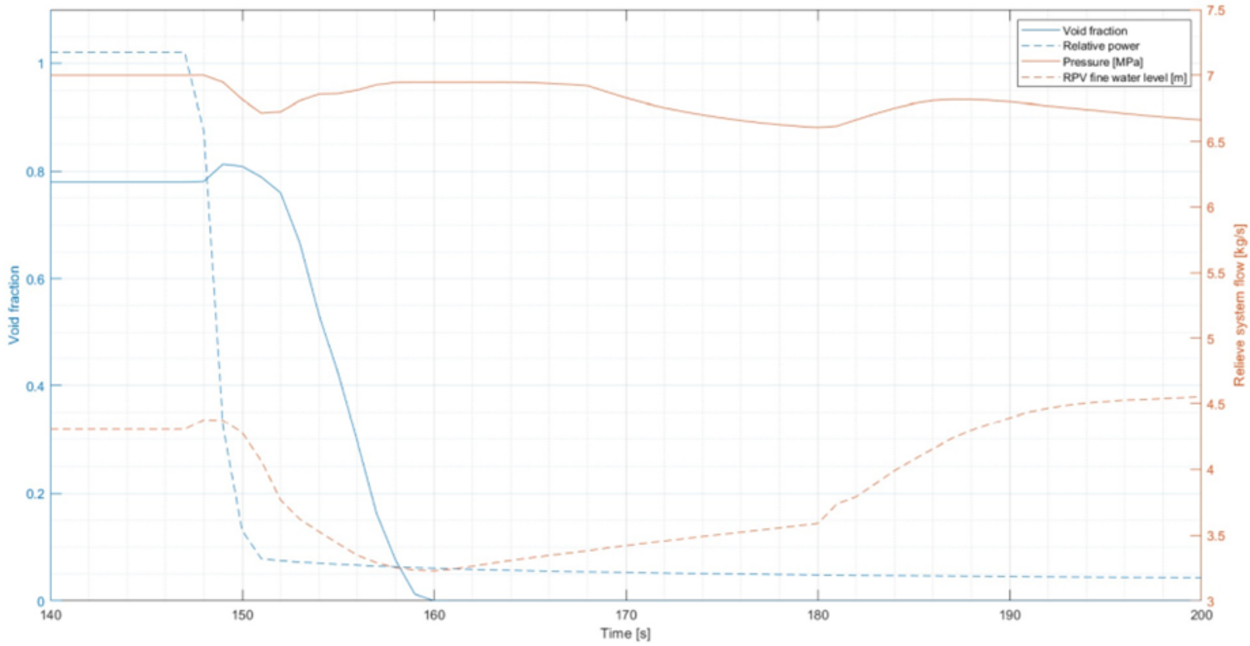

Figure 2.4: Void fraction, relative power, pressure, and water level

2.3 Interpretation

Control rod position

Due to the manually initiated SCRAM, the controlled rods were inserted into the reactor.

Power

When the control rods were lowered into the core during the manual SCRAM, it caused the absorption of neutrons responsible for fission. Consequently, the fission rate decreased, leading to a notable reduction in reactor power. This observation confirmed the validity of our initial prediction that the fission rate and reactor power would decrease during the manual SCRAM situation.

Pressure

The energy deposited into the coolant per unit of time decreased when the power decreased. So the coolant water did not boil as much as before. Thus, the steam production rate decreased at first. If the steam production rate decreased, the pressure decreased.

At around 149 seconds, the steam line isolation valves were closed, and the turbine trip happened at around 155 seconds. This meant that the turbine steam inlet valves were closed, blocking the steam flow to the turbine. Thus, the pressure increased.

The relief system flow started at around 167 seconds, causing the pressure to decrease significantly.

After that, at around 180 seconds, when the feedwater flow increased, more water was pushed into the

RPV, increasing the pressure. Finally, when the feedwater flow decreased, the pressure went down likewise.

Void fraction

The simulation results for the void fraction during the manual SCRAM situation partially confirmed the initial prediction. As the reactor power decreased due to the manual SCRAM, the void fraction in the core also decreased. However, initially, it spiked because of the closure of the steam isolation valves, causing the water-steam mixture to increase momentarily. The reduction in reactor power led to a decrease in the boiling of water in the core, aligning with the predicted behavior. Fig. 2.3 illustrates the graph of the void fraction over time, clearly showing the observed decreasing trend before the initial increase.

Main recirculation flow rate

The main recirculation flow rate simulation results aligned with the prediction based on the recirculation flow control system (RFCS) behavior. As the reactor power decreased during the manual SCRAM, the RFCS sensed the reduced heat production and slowed the recirculation pump. This logic responded to the decreased need for core cooling as less heat was generated. Consequently, the main recirculation flow rate simulation graph decreased over time, validating the prediction. Fig. 2.1 illustrates the correlation between the decrease in reactor power and the reduction in the main recirculation flow rate, confirming the role of the RFCS in regulating the recirculation system flow rate.

Steam flow rate

After the SCRAM, the reactor did not produce as much power. So the steam production rate decreased. With decreased steam production, the steam flow rate decreased as well.

Water level

The simulation results for the water level in the reactor pressure vessel (RPV) closely aligned with the predictions. The water level initially increased due to reduced evaporation after the manual scram. However, as feedwater flow decreased, the water level gradually decreased. The activation of the auxiliary feedwater system restored the water level, confirming its effectiveness in maintaining safety during the transient event. There was a slight bumpy increase during auxiliary feedwater flow (AFF) activation, corresponding to the peak of AFF as shown in Fig. 2.3. Moreover, the final water level was significantly higher than the starting water level because the pressure head had to use natural convection to cool the system, as the recirculation pump's influence was minimal.

Feedwater flow rate

The feedwater control system regulated feedwater flow to the reactor vessel to maintain the reactor water level. When the reactor water level initially rose, and the steam flow rate decreased, the feedwater control system sensed these changes and accordingly decreased the feedwater flow rate.

At around 180 seconds, the feedwater flow increased. Then it decreased again later. The increase could be an attempt to maintain reactor pressure above a certain threshold. However, we were not exactly sure about the explanation for this.

Auxiliary feedwater system (AFWS)

Because of the 3.1 m minimum water level requirement, the auxiliary feedwater system started, restoring the water level.

Relief system flow

The purpose of the relief system was to rapidly lower reactor pressure in an emergency. At around 167 seconds, the relief system flow depressed the RPV.

Chapter 3

Case 2- feed water enthalpy decrease

3.1 Description

In this case, we consider a scenario with a gradual decrease in feed water enthalpy during normal operation at nominal power. This situation is caused by the loss of feedwater preheaters, reducing the feedwater temperature from 185 °C to 161.5 °C over 60 seconds. This analysis aims to understand the system's response to such a feedwater enthalpy decrease and its impact on various parameters within the reactor.

3.2 Results

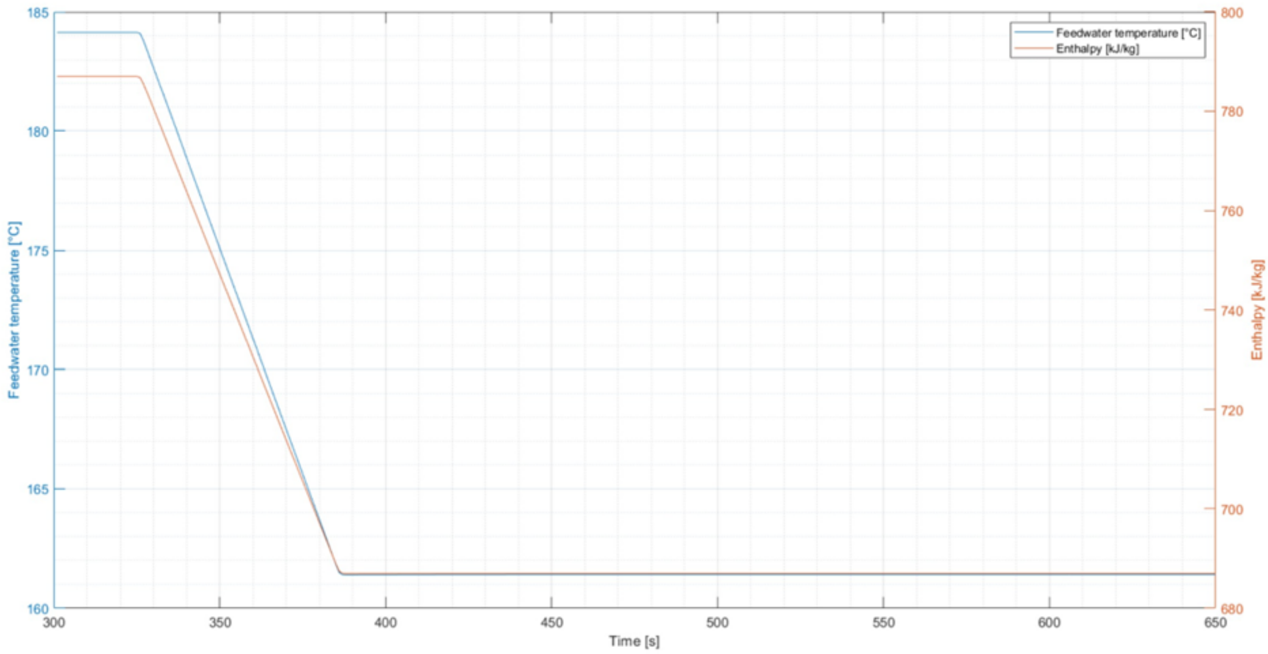

Figure 3.1: Feedwater temperature and enthalpy

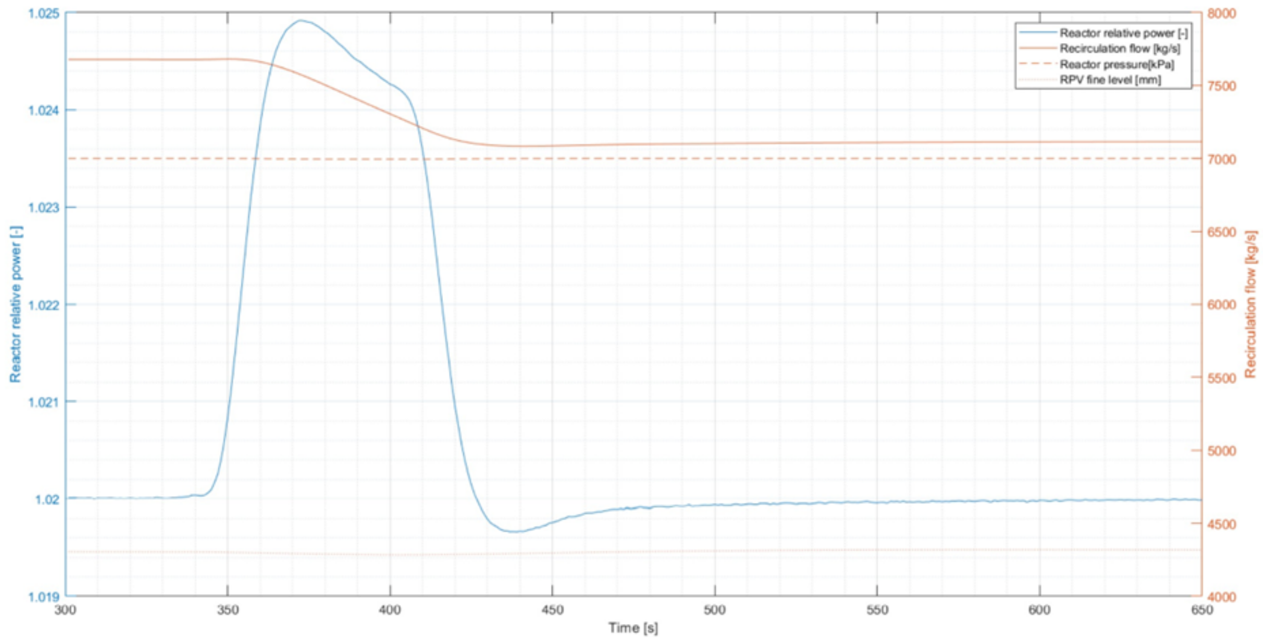

Figure 3.2: Reactor relative power, pressure, recirculation flow, and water level

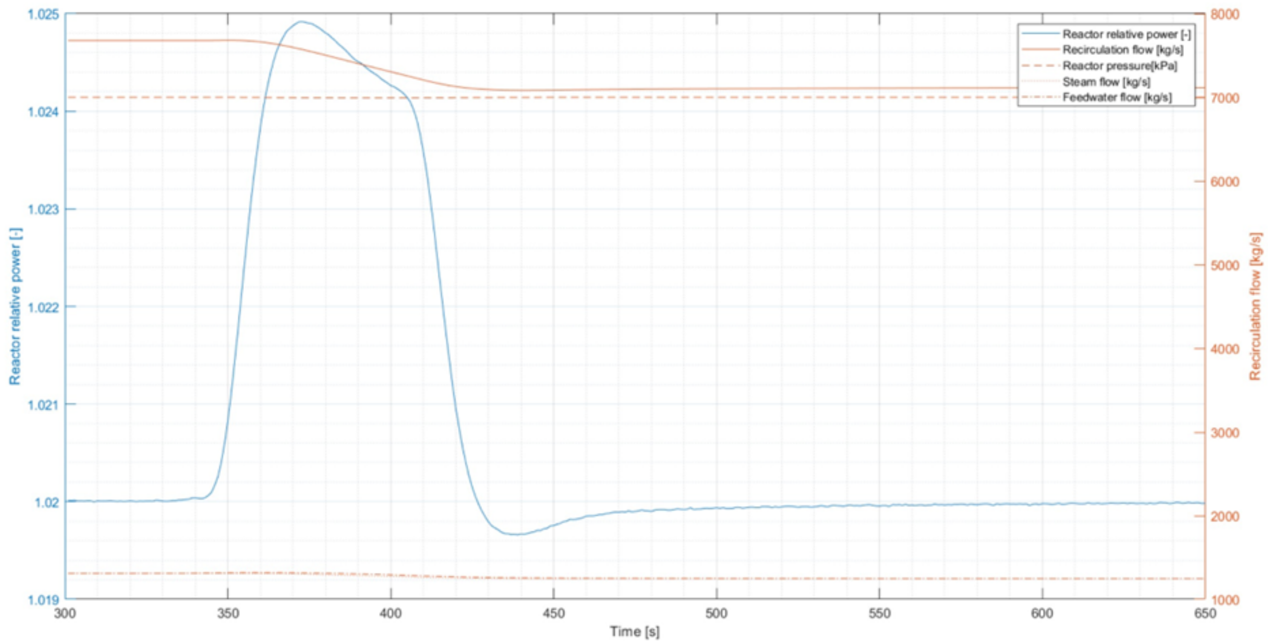

Figure 3.3: Reactor relative power, pressure, recirculation flow, steam flow, and feedwater flow

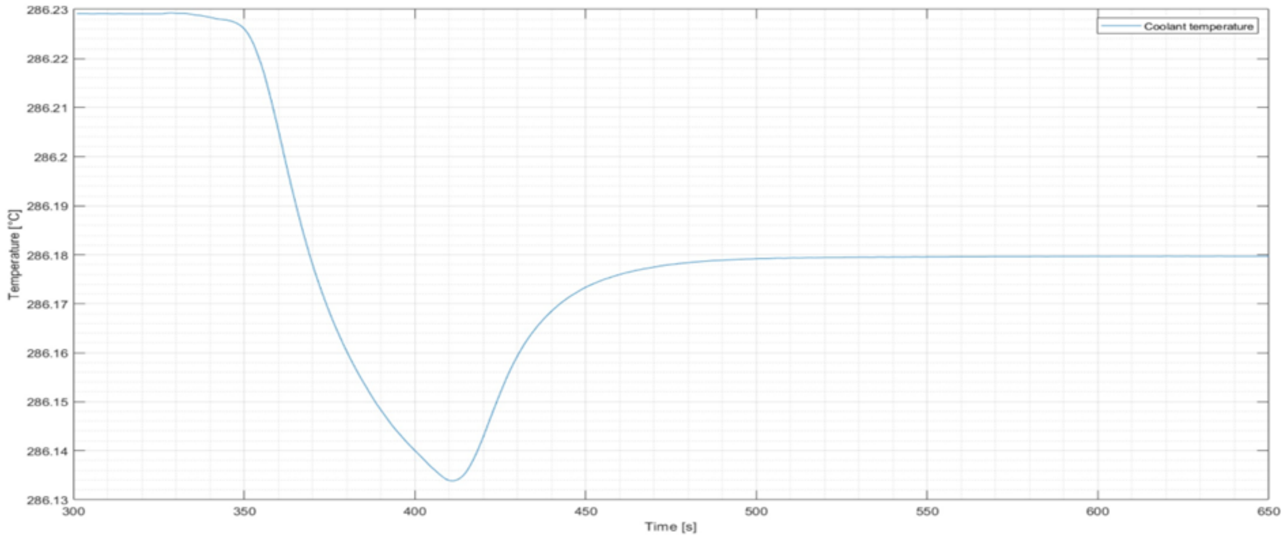

Figure 3.4: Coolant temperature

3.3 Interpretation

Feedwater temperature and enthalpy

The preheaters were lost. Thus, the feedwater temperature gradually decreased, and because of the decrease in feedwater temperature, the enthalpy of it also decreased.

Coolant temperature

When the feedwater preheaters were lost, the feedwater temperature decreased. This decrease was also reflected in the coolant temperature.

After some time, the coolant temperature increased again. We, unfortunately, could not explain this adequately. But we could make a conjecture that when the auxiliary feedwater system started, the slightly warmer water was injected into the core, and the temperature of the coolant rose a little again. But here we talked about a 0.05°C increase, which was insignificant.

Power

When the feedwater preheater was not functioning, the feedwater coming into the RPV was colder than normal. Due to the negative coolant temperature coefficient, the negative change in coolant temperature caused a positive change in reactivity. This increased the power of the reactor initially.

But when the reactor power increased, the recirculation pump was slowed down by the recirculation flow control system to control the rise in power. This reduced the power for some time.

Afterward, the coolant temperature rose again, which introduced negative reactivity in the system, resulting in a sharp decrease in power.

The power reached a new set-point as the recirculation flow became constant, and the coolant temperature reached a new steady state.

Main recirculation flow rate

The recirculation flow control system (RFCS) was responsible for controlling the speed of the recirculation pump to control the reactor power. So when the reactor power increased due to the coolant temperature decrease, the recirculation pump was slowed down. So the recirculation flow rate decreased.

Steam flow rate

There was a sharp but minimal (0.5%) increase in reactor power, but it was immediately met with a reduction in recirculation flow. So the reactor was trying to boil off more water but was not getting as much water in the core. So overall, the steam production rate stayed constant. Thus, the steam flow rate was constant.

Void fraction

We could not see the graph of the void fraction. So we could not interpret its behavior decisively. But since the reactor power increased and the recirculation flow decreased, the reactor tried to boil off more water, but less water was flowing into the core. So we could guess that the void fraction might increase slightly.

Water level

The reactor's water level stayed constant since the feedwater and steam flow rates were constant.

Feedwater flow rate

The feedwater control system regulated feedwater flow to the reactor vessel to maintain the reactor water level. It tried to imitate the steam flow rate and checked the water level to determine the feedwater flow rate. Since the steam flow rate was constant, the feedwater flow rate was also constant.

Pressure

The steam flow rate was constant. On the other side, the feedwater flow rate and the water level also stayed constant. So ultimately, the pressure inside the reactor did not change.

Auxiliary feedwater system (AFWS)

We could not see the graph of the auxiliary feedwater system. So we could not interpret its behavior. However, to explain the increase in the coolant temperature, we could assume that this system started.

When the auxiliary feedwater system started, slightly warmer water was injected into the core, and the temperature of the coolant rose a little again. But again, we were talking about a 0.05°C increase, so it was not much in magnitude.

Automatic depressurization system (ADS)

We could not see the graph of the automatic depressurization system. So we could not interpret its behavior decisively. But since the reactor pressure did not change, we could assume that this system did not start.

Chapter 4

Conclusion

Overall, we made predictions for the two scenarios that matched the results obtained from simulations in many cases. However, in some cases, for example, the predictions regarding the reactor pressure in case 2, or the predictions involving the fluctuating pressure in case 1, we failed to make correct predictions. However, we attempted to interpret the reason for these discrepancies according to the results of the simulations. There are some parts that we are not being able to describe from the understanding that we have of the BWR working principle. All in all, we gained great insight into transient cases for BWR.