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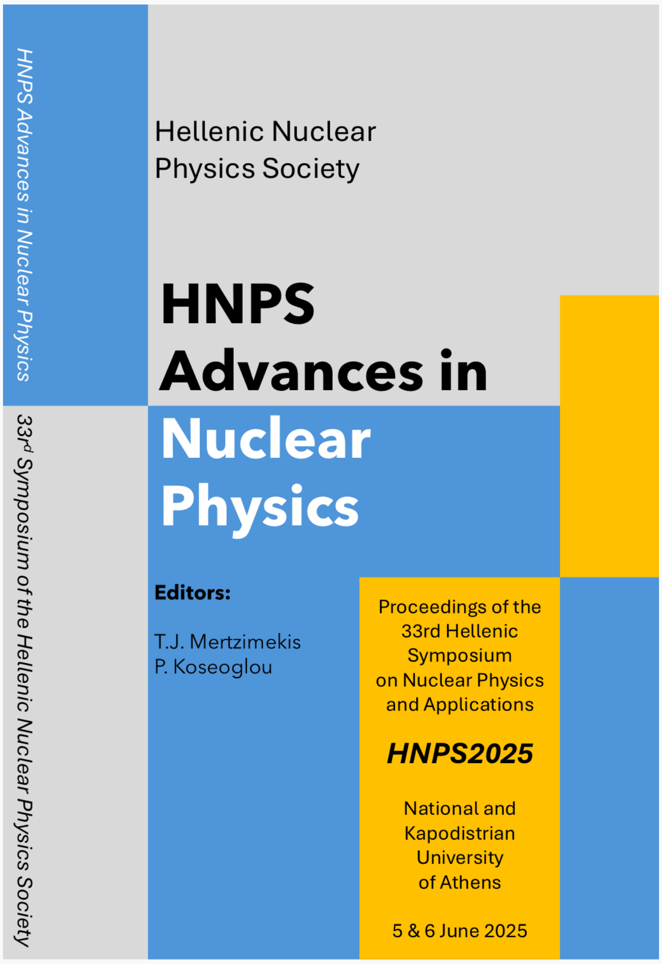

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ARTICLE

Analysis of iPWR SMR behavior during malfunction events

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Abstract

Small modular reactors are nuclear reactors with a maximum power capacity of 300 MWe according to IAEA's classification of nuclear reactors [1]. Advertised for their short installation time, compared to traditional Nuclear Power Plants, and their negligible CO_2 emissions, SMRs can act as a complementary power source to a renewable and carbon–free energy production power grid. Due to their low power and their passive safety systems, in case of operation failure, damage is minimized without human intervention. The shut down systems rely on physical phenomena to operate and minimize the risk of radioactivity release. For example, the passive decay heat removal system (PDHR) relies on natural circulation of the coolant, while the gravity driven water injection system (GIS) and the control rod actuation in case of SCRAM event rely on gravity.

This study exploits IAEA's iPWR simulation to observe the behavior of an SMR during different malfunction events. During malfunction events, different safety systems, such as the PDHR and GIS, are activated, as well as other safety systems such as the automatic depressurization system (ADS) and the pressure injection system (PIS), to prevent further damage. The malfunction events studied are turbine malfunction, loss of feedwater-flow, steam-line break inside containment building, and station blackout. In these scenarios, phenomena such as Xe build-up and decay heat from fission products after shutdown are analyzed. For example, in the loss of feedwater-flow malfunction, Xe reactivity reaches its most negative value of -3445 pcm after approximately 7 hours, while the average coolant temperature stabilizes at 155.5°C after 16 hours. Such results are valuable for the appropriate selection of materials and components of SMRs. Also, this study suggest an improvement in the given SMR design in the blackout malfunction event, in which the SMR cannot cool down properly since the ADS valves are "fail-close" valves and cannot depressurize the system, halting the operation of other safety systems. With suggestion of the "fail-open" design of these valves, this problem could be solved and further damage would be prevented.

Keywords: SMR; Safety; iPWR; simulation

1. Introduction

As countries try to meet the terms and objectives of the Paris agreement, interest around nuclear energy and SMR development has surfaced in an attempt to reduce greenhouse emissions and provide a stable energy source to the electrical grid. According to IAEA, reactors can be divided into 4 categories depending on their electrical power output [2]: Large conventional reactors, Medium sized reactors, Small modular reactors and Micro reactors, with a power output of 700+ MWe, 300-700 MWe, up to 300 MWe and up to 10 MWe, respectively. Small modular reactors (SMRs) are nuclear reactors that have a power capacity of up to 300 MWe output per unit (small) and can be fabricated in a factory and assembled on site (modular). After the Fukushima Daiichi accident, the nuclear power industry has been devoting increased attention to developing passive safety features and systems to prevent similar accidents [3]. For this reason, proposed SMR designs are equipped with passive safety systems in order to eliminate risk of radioactive release without the need of human intervention [1].

1.1 Reactivity control and impairments in PWRs

Reactivity in PWRs is impaired by changes in Xe concentration and temperature changes of the fuel and water coolant [4]. Xenon is the most important product poison due to its high absorption cross-section of thermal neutrons. It can be produced either by the decay of I-135 or directly by fission. During a decrease in power level, neutron flux is reduced (reducing Xe burnup), while Xe production from β^- decay of I-135 continues, causing an overall increase of Xe concentration that acts as a reactor poison, leading to increased neutron absorption [5]. Fuel temperature and water temperature have a negative feedback effect on reactivity. As fuel temperature rises, neutron capture cross-section of U-238 also increases, since the relative velocity between neutrons and nuclei increases at higher temperatures. This way, the resonance absorption line widens and covers a larger energy range. When a neutron is absorbed by U-238, it turns to U-239 which decays to Pu-239. This decreases the multiplication factor and the criticality of the system since more neutrons are absorbed for non-fission processes. As water temperature increases, density decreases and pressure increases leading to the ejection of water. Less water causes less moderation, thus lowering the number of thermal neutrons in the reactor core. The reverse happens in the case of temperature decrease. Also, during water temperature increase, the void fraction rises, meaning that air pockets could form in the mixture. In the air pockets, moderation of neutrons is significantly lower, leading to a decrease of thermal neutron flux [5, 6]. To counteract these effects, two methods are used are used to control reactivity. Firstly, control rods (movable pieces of neutron-absorbing material) are used. As the control rods are withdrawn or inserted from a critical reactor, the reactor tends to become supercritical or subcritical respectively. Also, a boric acid is dissolved in the coolant water, contributing to reactivity control [4, 5].

1.2 Related work

In previous work, multiple researchers have tested the reliability and capability of the safety systems of different gen II+/III+ models using various simulators. Zhao et al used the RELAP5/MOD3.4 code to simulate the passive residual heat removal system (PRHRS) of the IP100 reactor. The results indicated that the safety mechanisms combined with the PRHRS showed impeccable safety performance during loss of feedwater (LOFW) accident [7]. Similarly, Guo et al used ASTEC severe accident code to study the emergency core cooling system and the role of the containment vessel in the NuScale iPWR during loss of coolant accident (LOCA). The study showcased the ability of smaller vessels to minimize core fluid vaporization utilizing higher system pressure and water levels, and at the same time the ability of a similar design to manage the increase in nominal power from 160 MW to 250 MW [8]. In this simulation, IAEA's iPWR SMR simulator is used to highlight the safety systems of the reactor and its response to various severe accident scenarios like spurious turbine trip, loss of feedwater flow, steam line break inside containment building and station blackout

1.3 iPWR - General description

iPWR stands for integral Pressurized Water Reactor, and its main difference from a common PWR is the primary circuit. In the iPWR design, the primary circuit pipework components are placed within the reactor pressure vessel. The innovative iPWR design reduces the number of design-based accidents and offers possibilities that can be combined with electricity generation, such as thermal energy and medical isotopes [9]. However, in addition to their safety advantages, iPWR SMRs come with their own safety-related challenges due to the limited experience-based reliability data [10].

In this work, the integral Pressurized Water Reactor (iPWR) developed by IAEA and TECNATOM simulator was used to simulate different accident scenarios. This simulation is a part of the IAEA program in nuclear reactor simulation to assist Member States in educating and training nuclear professionals in the operation and behavior of nuclear power plants. Although not able to simulate reactor physics and thermal hydraulics phenomena as accurately as other simulators such as serpent or relap, the iPWR simulator acts as a reliable educational tool that can be used in computers even with limited capabilities and does not need as much time by the user to comprehend, while the simulation times are also much lower [11].

2. Materials and Methods

2.1 iPWR - Simulated systems

In the present work, the iPWR developed by IAEA and TECNATOM is used to investigate the behavior of the reactor during the following malfunction events: Spurious turbine trip, loss of feedwater flow, steam-line break inside containment building and station blackout. The following systems are simulated:

1. Reactor coolant and reactor core (RCS),
2. main steam system (MSS),
3. feedwater system (FWS),
4. turbine system (TUR),
5. generator system (GEN),
6. condenser system (CNR),
7. circulating water system (CWS),
8. containment building system (CBS),
9. automatic depressurization system (ADS),
10. containment cooling system (CCS),
11. gravity driven water injection system (GIS),
12. pressure injection system (PIS),
13. passive decay heat removal system (PDHR),
14. protection and control system (PCS).

The reactor is designed to produce 150 MW(th) using UO_2 at 4.95% enrichment. All systems are equipped with appropriate transmitters and probes placed in fixed positions to provide sufficient information about the reactor's state, to help the operator monitor and control the simulation. For example, the RCS is equipped with a pressure transmitter and temperature transmitters in the inlet and the outlet of the reactor core to properly monitor the core's state. A simplified version of the simulated iPWR is shown in Fig. 1. The simulator provides certain choices to the operator regarding the operation and simulation of the reactor. Firstly, the user can decide between two operating modes: Turbine leading (in which the turbine valve controls generator power) and reactor leading mode (in which the turbine valve and the turbine bypass valve control the steam header pressure). Also, the user has the ability to choose between natural or forced coolant circulation inside the core. Only natural coolant circulation will be used for the purpose of this research. In this case, no reactor coolant pumps are used, and the coolant flows naturally inside the core. In addition, different periods of the fuel's life cycle can be chosen for studying via the initial conditions of the simulation. The user has three choices: Beginning of life (BOL), middle of life (MOL), and end of life (EOL). During different parts of the fuel's life, different boron concentrations are used to compensate for fuel's depletion. For the purpose of this research only fuel in the beginning of life is used.

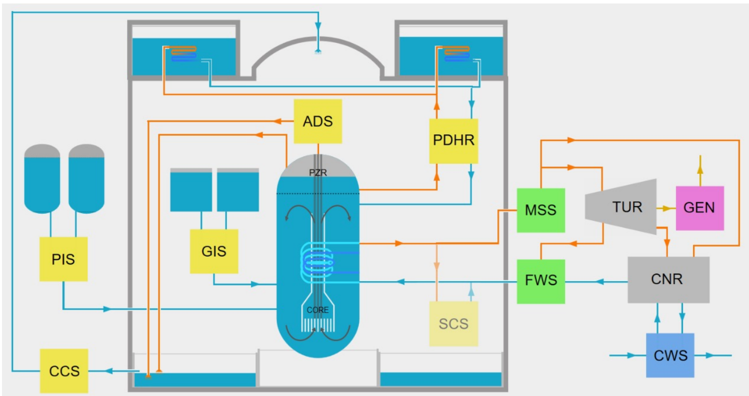

Figure 1. iPWR simulator overview

Lastly, the simulation gives the opportunity to simulate multiple generic and specific malfunctions. Generic malfunctions are considered all malfunctions regarding pump, valve or transmitter operation, and pipe rapture of a heat exchanger, while specific malfunctions are more complex scenarios like station blackout (SBO), spurious turbine trip or seismic event. In this study, the following malfunctions will be simulated to study the reactor's operating systems: Spurious turbine trip, loss of feedwater flow, steam line break inside containment building, and station blackout.

3. Results and Discussion

3.1 Spurious turbine trip

During this malfunction scenario, a spurious turbine trip and the subsequent recovery were simulated. During the turbine trip, it is important to maintain water circulation through the core to prevent overheating and damage in the core. At the same time, a full shutdown is not optimal since there will be a lot of Xenon accumulation and the startup operation will be harder. For these reasons, a reactor stepback at 60% (90 MWth) is preferred. Since the turbine is tripped, the steam is redirected directly to the condenser via the turbine bypass valve, this way constant water circulation is ensured. This operation procedure can be useful in the event of turbine/generator malfunction and maintenance. As the turbine trip is initiated, the reactor stepback to 60% is activated. After the turbine trip is reset, the reactor generator is restored to full power. In Fig. 2, the turbine valve closes after the turbine trip is initiated and the bypass valve opens to redirect heat to the condenser.

The boron concentration is raised by the operator and the control rods are lowered automatically as the thermal power lowers, in an attempt to ensure proper neutron flux and fuel burnup throughout the fuel. After the trip is reset and the power of the reactor is increased, the boron concentration is adjusted again. In Fig. 3 we can clearly see that after bringing the reactor back to full power, the boron

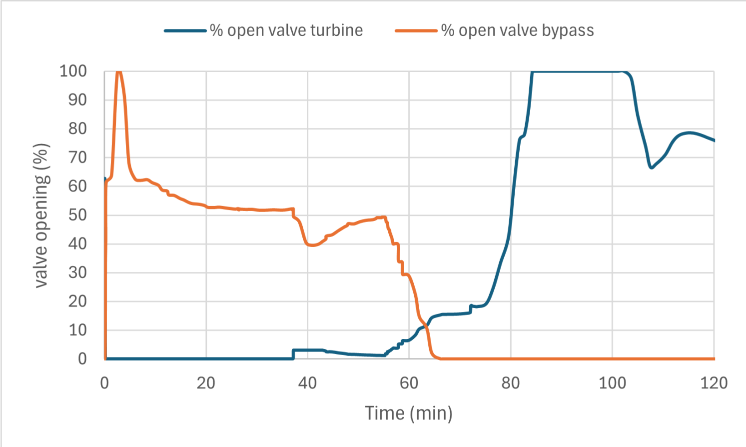

Figure 2. Turbine valve - Bypass valve opening (0-120 min)

concentration is lower than initially (764 ppm initially, 720 after the trip reset). That is caused to due Xe buildup which reached a minimum of ~ -2581 pcm. After approximately 12 hours,equilibrium is restored and the Xe reactivity stabilizes at ~ -2338 pcm.

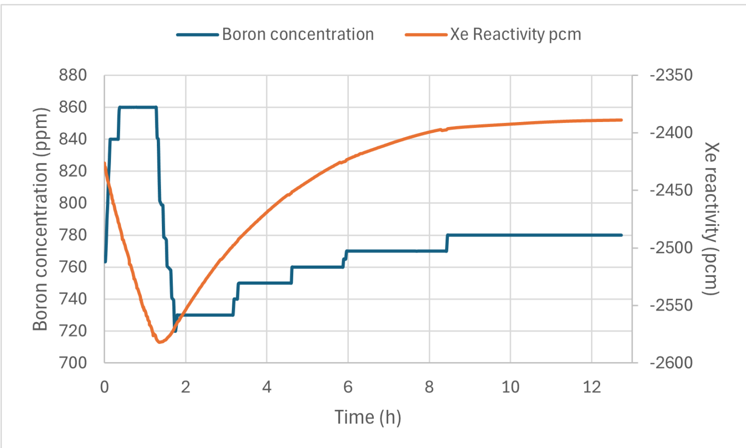

Figure 3. Boron concentration (ppm) - Xe reactivity (pcm) (0-12 hours)

As shown in the graphs, the power plant's operating systems were able to handle the accident without the need for complete shutdown of the reactor or actuation of emergency systems. The reactor is brought back to full power with minimized Xe buildup.

3.2 Loss of feedwater flow

In this malfunction scenario, we simulate an FWS malfunction. With no water circulation, a reactor trip is inevitable. Cooling has to be maintained throughout the core to remove decay heat of fission products from the core and keep the fuel intact. This heat will be redirected to the PDHR pools via natural circulation. In a similar work, Zhao et al investigate via simulations the capabilities of the passive safety systems of an iPWR, exploiting natural circulation, to control this type of malfunction [7]. In our simulation, activation of ADS and PDHR is expected, which is then followed by the activation of PIS and GIS.

Initially, the FW pumps are tripped, stopping the FWS operation. The water circulation is stopped and thus, the heat removal system. The FW pumps trip causes reactor and generator trip. All controls immediately drop inside the core, and neutron flux drops to 0. Due to loss of primary heat sink, PDHR is automatically activated and water begins to circulate through the PDHR pools. ADS is activated to depressurize the RPV, causing the activation of PIS initially and then GIS. The water flow of the systems mentioned can be seen in Fig. 4. The coolant temperature stabilizes after 16 hours at 115°C while the Xenon reactivity reaches its most negative value of -3445 pcm after 7 hours. In Fig. 5 it is shown that after the depletion of PIS and GIS, the coolant temperature rises due to the decay heat from the fission products. This heat is dissipated to the PDHR.

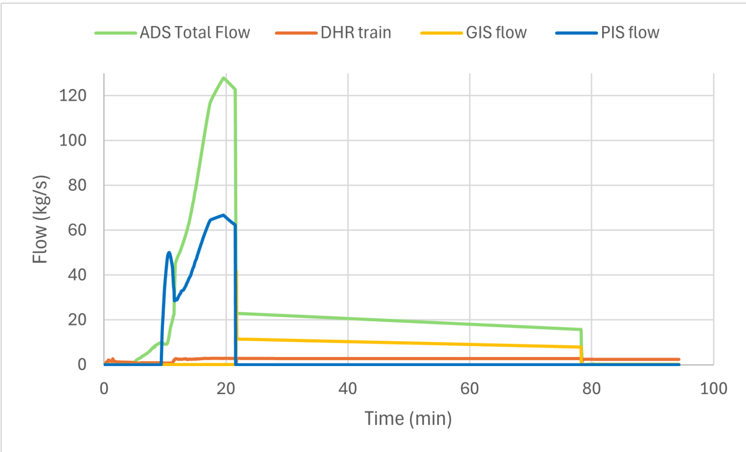

Figure 4. ADS, PIS, GIS, PDHR flow (kg/s) (0-100 minutes)

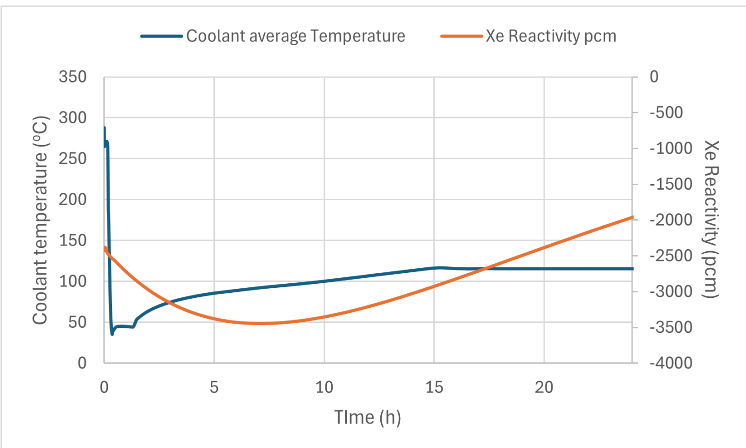

Figure 5. Coolant temperature (°C) - Xe reactivity (pcm)

The operator has the ability to deactivate ADS and not depressurize the core [7], in this case, the pressure is not properly decreased, and PIS and GIS are not activated since low pressure is needed for their automatic activation. Heat will be removed through natural circulation of coolant water to the PDHR pools. These pools contain a large volume of cold water and heat exchangers capable of removing heat from the core. A similar scenario where ADS activation is not applied will be seen in detail in 3.4.

3.3 Steam line break inside containment building

During this malfunction scenario, a steam-line break inside the CBS building is simulated. After a steam-line break, pressure in the CBS increases. To minimize the chances of radiation leakage to the environment or even possible explosion due to hydrogen accumulation, immediate shutdown of the core and depressurization of the CBS is necessary [12]. As a result of the malfunction, the FW pumps are tripped in order to stop water circulation and therefore further leakage in the CBS. This means that the reactor and turbine are tripped as well. There is loss of heat sink and the main heat removal system is halted. Since water circulation is stopped, the PDHR is automatically activated.

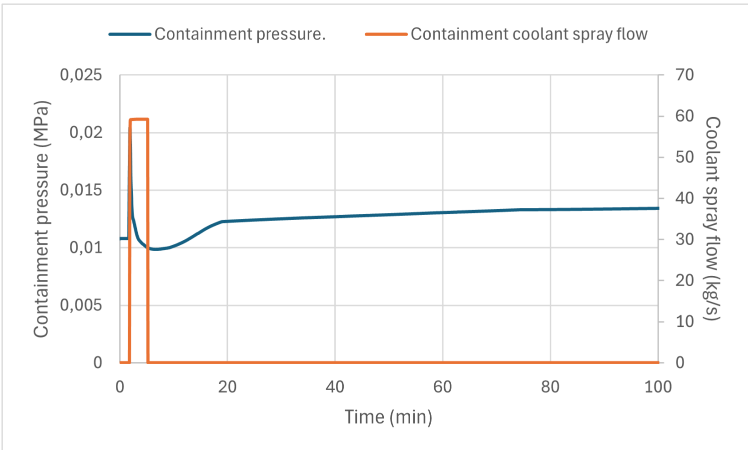

Figure 6. Containment pressure (MPa) - CCS spray flow (kg/s)

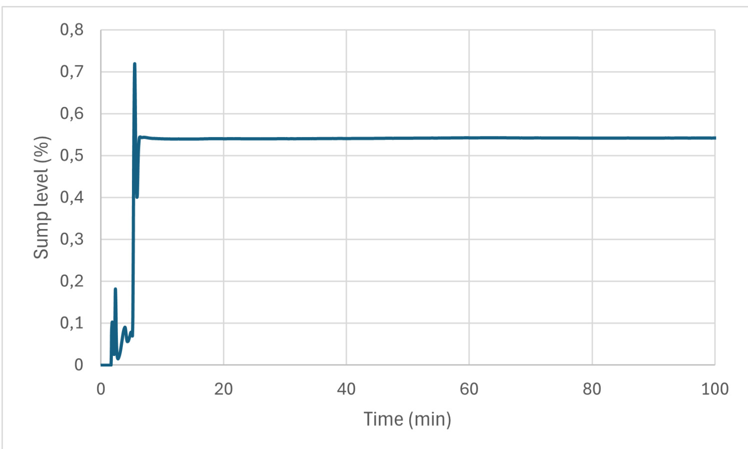

Figure 7. Water level in sump(%)

The main difference between this malfunction and the one mentioned in section 3.2 is that due to steam-line break, the CBS pressure rises. This causes the containment coolant spray to activate in order to cool down and reduce pressure inside the CBS. A sump below the RPV is responsible to collect any leaking water from the RPV in case of accident. So, in this scenario, since coolant leaks from the RPV and accumulates in the sump pool, the water level rises.

The rise of the containment pressure and the activation of the CCS can be seen in Fig. 6. After approximately 70 minutes, the containment pressure stabilized at 0.0134 MPa. The rise of the sump level can be seen in Fig. 7. The sump level stabilizes at 0.54% after the 7th minute. After successful activation of the safety systems, CBS is kept in acceptable temperature and pressure values, ensuring no further damage to the plant and radiation leakage to the atmosphere.

3.4 Station blackout

During the last malfunction scenario, we simulate a total station blackout. In similar simulations, we can see that iPWRs are capable of removing the residual heat from the core [13]. During our simulation, the loss of power halts all water circulation and heat sink is lost. The FW pumps stop working and the heat removal system is non-respondent. PDHR is automatically actuated since heat sink is lost. The control rods are gravity actuated and instantly drop in the core after loss of power. The reactor is shutdown and the generator is tripped. This time, ADS does not depressurize the RPV since the ADS valves are fail-close valves. The RPV pressure does not drop properly and PIS is only partially activated, while GIS is not activated at all. More specifically, the RPV pressure drops from 15.5 MPa to 4.5 MPa and PIS slowly injects borated water until RPV and PIS tanks achieve pressure equilibrium. This phenomenon can be seen in Fig. 8.

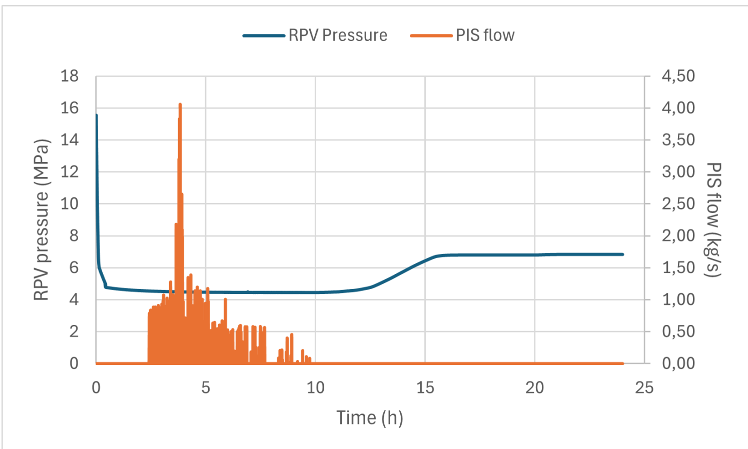

Figure 8. RPV Pressure (MPa) - PIS flow (kg/s)

As the PDHR pool temperature and the average coolant temperature slowly achieve temperature equilibrium, the PDHR flow is reduced, increasing the RPV pressure to 6.8 MPa after 15 hours. The average temperature of the coolant stabilizes at 128°C (+13°C compared to section 3.2 where safety systems operated at maximum capacity), and the PDHR pool temperature at 108°C. This can be seen clearly in Fig. 9.

This increase in temperature could have been avoided by proper depressurization of the RPV. In order for that to happen, fail-open valves could be used to ensure proper depressurization in case of station blackout. However, the effectiveness of this change needs to be verified using more advanced simulation tools with enhanced capabilities.

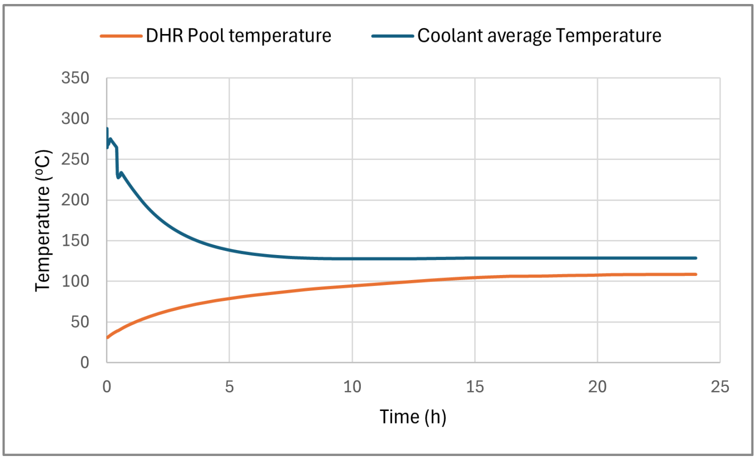

Figure 9. Average coolant temperature - PDHR pool temperature (°C)

4. Conclusion

During the simulation, we were able to observe and analyze the performance of the reactor safety systems while demonstrating the training capabilities and educational value of the simulator. During the spurious turbine trip, the reactor stepback successfully limited xenon buildup. In the subsequent malfunction scenarios, where all safety systems operated correctly, the SCRAM procedure was executed automatically without the need for operator intervention, leading to a controlled decrease in reactor's temperature and preventing further plant damage. This behavior was evident in both the loss of FW flow and steam line break simulations. In these cases, once the safety systems engaged, the temperature returned to acceptable levels. The only significant difference between the two scenarios was the activation of the CWS during the steam-line break accident, which helped reduce pressure buildup in the containment building. The station blackout scenario was the exception. Here, fail-close valves prevented the proper shutdown sequence, resulting in a slower temperature decline until reaching equilibrium. Although further investigation is required, our data suggests that by replacing the fail-close valves with fail-open design, would allow the SCRAM sequence to proceed as intended.

Acknowledgements

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