SENSITIVITY OF INTEGRAL REACTOR CONFIGURATIONS TO PARAMETRIC FLUCTUATIONS

A Thesis

by

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Submitted to the Graduate and Professional School of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

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May 2023

Major Subject: Nuclear Engineering

ABSTRACT

Since the 2019 Climate Change and the Role of the Nuclear Power International Conference and the Paris agreement goal, the international community started to be interested in nuclear power. Small modular reactors (SMRs) can reduce global warming effects and help to achieve net zero emission objectives. The International Atomic Energy Agency (IAEA) collected SMR design data for several years. Compared to conventional reactors, small modular reactors usually connect to micro or mini electricity grids, which are more sensitive to the reactor parameters' fluctuations. Therefore, this research analyzes the small modular reactor's oscillations and performances via the IAEA small modular reactor simulators. The analysis is performed using the IAEA i-PWR simulator.

The scenario plots are in reference to the regular nominal nuclear power plant operations, such as base-load and load-following modes. A range of operational parameters have been explored such as normalized power levels of 100%, 75%, 50%, and 25% in transients for 6, 12, 24, 48, and 72 hours. The reactor transient performance shows that the fluctuations of normalized power are related to the xenon-135 concentration and control rod worth's oscillations. The lower power level cases have more sensitivity to the xenon concentration fluctuations. The trend of the lower power level, except for 25% cases, shows delayed behavior. The load-following scenarios for normalized power and generator load correspond to the base-load mode. The deviation variations do not change significantly, whether base-load or load-following are selected. However, when increasing

the operation time in the base-load mode, the standard deviation of normalized power and generator load may increase or decrease depending on the power level of operations.

DEDICATION

To my parents and my brother who always supported me, to my friends and colleagues who motivated and encouraged me to complete this thesis.

ACKNOWLEDGMENTS

I would like to huge thanks to my mentor and committee chair, Dr. Tsvetkov for his guidance and encouragement on this thesis work. Without his useful advice and high efficiency in replying to my questions, this research could not have been completed. I would like to thank my committee members Dr. John Ford of the Department of Nuclear Engineering and Dr.Michael Pate of the Department of Mechanical Engineering for their academic advice.

Special thanks to my friends and colleagues for their recommendations to help me complete this research. Their support and comfort gave me enough energy to do this research. Finally, I want to thank the Department of Nuclear Engineering provides graduate assistantships for me and the faculty members who teach me during my master's life.

CONTRIBUTORS AND FUNDING SOURCES

Contributors

This work was supported by a thesis of consisting of Dr. Pavel Tsvetkov [advisor and committee chair] and Dr. John Ford of the Department of Nuclear Engineering and Dr. Michael Pate of the Department of Mechanical Engineering. The simulator used for this research was provided by the IAEA to member states. All other work conducted for this thesis was completed by the student independently.

Funding Sources

Graduate study was supported by Graduate Teaching Assistantship from the department of Nuclear Engineering, Texas A&M University.

NOMENCLATURE

ADS	Automatic Depressurization System
BOL	Beginning of Life
CNR	Condenser System
DOE	Department of Energy
EPZ	Emergency Plan Zone
EUR	European Utilities Requirements
FC	Forced Circulation
FWS	Feedwater System
GEN	Generator System
GIS	Gravity-Driven Water Injection System
i-PWR	Integrated-Pressurized Water Reactor
IAEA	International Atomic Energy Agency
LOCAs	Loss of Cooling Accidents
MOL	Middle of Life
MSS	Main Steam System
MW	Mega-Watts

MWe	Mega-watts electric
MWth	Mega-watts thermal
NC	Natural Circulation
NRC	Nuclear Regulation Commission
NSSS	Nuclear steam supply system
PCS	Protection and Control System
PCTRAN	Personal Computer Transient Analyzer
PDHR	Passive Decay Heat Removal System
PIS	Pressure Injection System
PWR	Pressured Water Reactor
PZR	Pressurizer
RCS	Reactor Coolant System
SMR	Small Modular Reactor
TRAC_RT	Transient Reactor Analysis Code - Real time
TUR	Turbine System
Xe	Xenon

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1. INTRODUCTION

To achieve the Paris agreement goal, scientists and engineers need to recognize all low-carbon sources of energy to limit the rise in global temperatures. Besides, the public is aware and concerned about the severe impacts of global warming such as rising sea levels and droughts. Thus, the electricity generation ratio of renewable energy, including solar power and wind power continues to grow and gradually replaces fossil fuels. However, those renewable energy sources are unlike nuclear power which can provide steady and reliable electricity that will affect countries' economies, especially developed countries. In the 2019 International Conference on Climate change and the Role of Nuclear power, many countries began to increase interest and consider small modular reactors (SMRs) as a potential nuclear option to alleviate global warming. [1]

Currently, according to the International Atomic Energy Agency (IAEA) data, there are at least 70 SMR designs under development for different applications. [1] Compared to 2018, it increased by approximately 40%. Now SMRs have become a global trend. For instance, Argentina CAREM, an integral pressurized water reactor and China HTR-PM, a high-temperature gas-cooled reactor plan to operate between 2021 and 2023. [2, 3] In May 2020, Russian (KLT40s, a floating power unit) started commercial operation and connected to the electricity grid. [1] The United States (NuScale, an integral pressurized water reactor) expects to construct the first full-scale NuScale power plant in 2023 and predict commercial operations in 2027. [4]

Generally, this research will focus on the electricity grid fluctuation of SMRs since SMRs have become more popular; however, the parameters data usually borrow from pressurized water reactors (PWRs), so it is necessary to explore and simulate for SMRs. In addition, PWR usually connects to a large electricity grid. Compared to SMRs, they work with a small-scale electricity grid. Thus, it is crucial to discuss and analyze this condition if we want to operate and commercialize SMRs in the future. Although now there are several different simulators for running, the IAEA simulator has not been discussed yet.

1.1. Research Objectives

The thesis research objectives mainly focus on the sensitivity study of transients in integrated PWR-based small modular reactor configurations. The goal of this study is to quantify sensitivities of observable balance of plant performance parameters to fluctuations in internal performance characteristics. The parameters include power distributions, normalized power levels, and external generator load fluctuations assuming nominal daily operation conditions. The research efforts account for dynamic and steady-state behavior characteristics.

1.2 Research Plan

The research is planned to be conducted using the IAEA i-PWR broad-scope plant simulator. [1] In order to meet the research objectives, the work was divided into the following tasks:

i. Assemble a set of performance characteristics capturing features of an integrated reactor unit during operation. Execute the i-PWR simulator accounting for the identified performance characteristics and analyze the results. Investigate the daily operation settings and set up series of transient scenarios accounting for reactor power demand fluctuations.
ii. Execute the scenarios in the IAEA i-PWR simulator
iii. Analyze the output data from the IAEA simulator using MATLAB or Microsoft excel to evaluate reactor transients.
iv. Analyze the sensitivity of input/output parametric combinations under different system conditions and configurations accounting for power levels fluctuations and time in operation.

1.3. Overview of Small Modular Reactors (SMRs)

The notion of SMRs came up in the 1950s to 1960s. It is developed for naval propulsion reactors and military purposes. [2, 3] During the 1980s, The U.S Department of Energy (DOE) claimed "new interest in small and medium-size reactors and in more advanced reactor concepts other than those marketed today." Meanwhile, IAEA began to publish serial studies and designs of SMR until today. [5] Since the 2010s, the DOE has invested over $1 billion to support several nuclear private companies such as NuScale, Bechtel, and so on for SMR setup and commercialization. This program and construction will be extended to 2025. [6]

An SMR is an advanced fission reactor whose electrical power level is up to 300-megawatt electric (MWe) as defined by IAEA. [1] Although SMRs have lower power levels compared to large conventional reactors they offer significant advantages accounting for their deployment flexibility and overall resilience. For example, the modular design of SMRs makes them easier to construct and transport to a remote location. Owing to these factors, building SMRs can reduce the cost and construction time more than conventional reactors. In general, the uniqueness of SMRs has fewer vessels and pumps making SMRs' design simpler. In addition, an SMR has inherent safety features or passive safety systems. With those components, it lessens the dependence on active safety systems that decreases the probability of lost cooling system accidents (LOCAs). Because the fuel size is small and the total amount of fuel pins is less than the conventional reactor, they reduce the storage demand for fuel and radioactive material. Many of SMRs are designed to be installed underground giving high protection to terrorist threats, as shown in figure 1. [6]

Figure 1. SMR configuration of NuScale Power Module developed by NuScale Power Inc., United States of America. Reprinted from [4]

In accordance with those characteristics including small size and less demand of fuel inventory, the SMRs' evacuation zone can be minimized and the emergency planning zone (EPZ) can be reduced to no more than 300 meters radius. [3] As opposed to conventional large nuclear power plants, the United State Nuclear Regulation Commission (NRC) sets EPZ to be 2 miles radius (3.21 kilometers). [7] And at the beginning of the Fukushima disaster, the Japanese government emergency evacuates the public far from the damaged power plant within a 30 kilometers radius. [8] Based on this disaster, the reduction range of EPZ and evacuation zone is an appealing advantage for SMRs.

SMRs can be built as single units or multi-units according to user needs. For instance, in an off-site remote or small population county, the requirement for electricity may be less. However, if the area has high growth of population, which increases the electricity demand, they do not need to rebuild the large reactors. They can extend units, and each unit can connect together becoming multi-units. The multi-units allow users flexibly adjust generating power. Furthermore, small-scale power generating lets SMRs not need to work with a large electricity grid. The benefits of replacing a large grid with a smaller grid are more cost-efficient, decreasing energy loss, and reducing carbon emissions because a small grid can operate and adjust in low load at night or based on the electricity demand. [5, 9]

IAEA categorizes SMRs into several different types such as Land-Based Water-Cooled SMRs, Marine Based Water Cooled SMRs, High-Temperature Gas-Cooled SMRs, Fast Neutron

Spectrum SMRs, Molten Salt SMRs, and Micro-sized SMRs. [1] In general, their fuel, coolant, and module material are determined by their reactor types and designs. In this thesis research, it will focus on the Land-Based Water Cooled SMRs. Table 1 summarizes the main properties and outlines of different countries' designs nowadays. To be clearer, this research particularly concentrates on integral-Pressurized Water Reactors (i-PWRs) simulation.

Table 1 Land-Based water cooled SMRs. Modified from [1]

	NuScale	BWRX-300	CAREM	ACP100	SMART	CANDU SMR	UK-SMR	NUWARD
Country	United States of America	United Stated of America and Japan	Argentina	China	South Korean and Saudia Arabia	Canada	United Kingdom	France
Design Organization(s)	NuScale Power, Inc.	GE Hitachi and Hitachi GE Nuclear Energy	CENA	CNNC/NPIC	KAERI, K.A. CARE	Candu Energy	Rolls Royce, Plc.	EDF
Reactor type	i-PWR	BWR	i-PWR	i-PWR	i-PWR	PHWR	3-loop PWR	i-PWR
Primary circulation	Natural circulation	Natural circulation	Natural circulation	Forced circulation	Forced circulation	Calandria	Forced Circulation	Forced circulation
Fuel type/ assembly array	UO_2 pellet/ 17*17 square	UO_2 pellet/ 10*10 array	UO_2 pellet/ hexagonal	UO_2/ 17*17 square	UO_2/ 17*17 square	37 element fuel bundle	UO_2/ 17*17 square	UO_2/ 17*17 square
Coolant	Light water	Light water	Heavy water	Light water	Light water	Light water	Light water	Light water
Moderator	Light water	Light water	Heavy water	Light water	Light water	Light water	Light water	Light water
Electrical output (MWe)	60 (gross)	270-290	30	125	107	30	443	2*170
Thermal power output (MWth)	200	870	100	385	365	100	1276	2*540
Reactor vessel's height/diameter (m)	17.7/2.7	26/4	11/3.2	10/3.5	18.6/6.5	N/A	11.3/4.5	13/4

1.4. Overview of Integral-Pressurized Water Reactors (i-PWR)

The first commercial i-PWR prototype, launched in 1964 used the German nuclear ship, Otto Hahn. In the 1980s, because of the Three Mile Island accident, many scientists tried to improve the safety of nuclear power plants. At that time US Combustion Engineering proposed the true i-PWR design that named it the minimum attention plant. After a few years, US Combustion Engineering revised this design and turned it into 320 MWe Safe Integral Reactors. Since the 1990s, Safe integral Reactors have become the typical i-PWR design and keep developing until now. [10]

The features of i-PWRs are minifying the vessel and pipes amount in order to decrease the coercion of tubes causing leakage. Beyond this, the construction designs of i-PWRs are different from Pressurized Water Reactors (PWRs). A typical i-PWR combines a steam generator, a pressurizer, and coolant pumps together. Figure 2 shows the configuration of PWR and i-PWR's discrepancy. In Figure 2, it can be noticed the pipe diameter is hugely decreasing the probability of leakage and eliminating the complex reactor vessel structures. [11] As we know several nuclear disasters happened such as the Fukushima and the Three Mile Island accident, the i-PWR design can help us fulfill the goal of LOCA reduction or mitigate the consequence of the emergency accident.

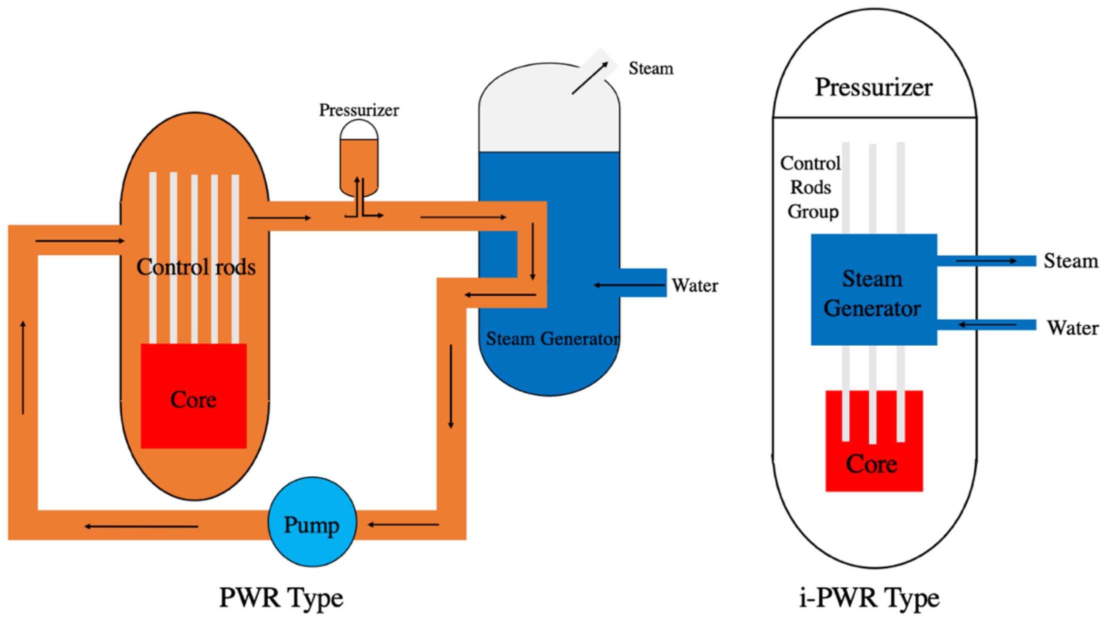

Figure 2. The main design differences between PWR (left) and i-PWR (right). Modified from [11]

The height of fuel pins in a PWR type is approximately twice as long as an i-PWR type. A PWR's average fuel cycle length is less than an i-PWR. However, the fuel cycle length is prone to the uranium enrichment, burnup ratio, operating time, etc. Hence, there is no absolutely that the PWR fuel cycle length is shorter than the i-PWR. In addition, the regular pressure of PWRs is about 2500 psi (17.2 MPa) which is larger than i-PWRs. Furthermore, the frameworks of PWRs' steam generations usually have two types. One is U-tube, and the other is one through to heat exchangers. Contrastingly, the i-PWRs have a U-tube or helical coil class. Helical coil steam generations can arise the heat transfer rate through the surface and compress the space of steam generation, which also decreases the thermal pressure on the feedwater and less vibration induced by the flow. As a result, most i-PWRs adopt the helical coil steam generations now, and fewer tubes connect to the steam generation. Moreover, i-PWR integrates the pressurizer located at the top of the reactor pressure vessel and coolant system that leads i-PWRs to have a larger volume of pressurizer and reactor coolant water inventory. With those factors, the transients of the pressure become slower, which gives the operators several benefits. For example, they may have more time to respond or to analyze the conditions, and the i-PWR system design provides water level in the reactor directly reducing human errors. The i-PWRs' natural circulating cooling system design lets i-PWRs have the ability to remove exceeding decay heat during shutdown causing the reactor coolant pump can be reduced. [10] Therefore, the i-PWRs design tries to solve PWR defects based on the experience of accidents. Essentially, it attempts to eliminate the coolant leakage and downgrade pipe broke situations by getting rid of complicated vessel design.

2. ANALYZE APPROACHES USING i-PWR SIMULATOR

Numerous computational techniques and models have been developed to provide capabilities to capture nuclear reactor behavior characteristics as a function of time accounting for operational features either in real-time or closely following real-time behaviors. Section 2 will briefly discuss the previous research works using other simulators relative to the IAEA i-PWR simulators' tools. A description of the IAEA simulator will be given as it is used in the present effort. Section 2.1 will introduce other simulators such as Modelica code-based models, Fortran-based models, and PC-based PCTRAN models. The next section 2.2 will illustrate the detail of the IAEA simulator and how to utilize it for this research. Section 2.3 summarizes general views of the i-PWR simulators that had been discussed.

2.1 Broad-scope simulators and their use in system evaluations

The Oak national Laboratory (ORNL) and Idaho national laboratory (INL) developed the i-PWR type reactor via Modelica code and Fortran. They use Simulink to connect each part of the simulator system such as the reactor core, pressurizer, control system, etc. The micro-simulation technology company developed the PCTRAN model based on the different vendor designs to establish the specific i-PWR reactor simulators' products. The following section will describe those simulators respectively.

2.1.1 Modelica Code Based Model

The Oak National Laboratory (ORNL) and Idaho National Laboratory (INL) used Modelica modeling programming language to create i-PWR simulators and devised the dynamic models for reactor subsystems. For Oak National Laboratory, they built 160 Megawatts thermal (MWth) and 53 MWe generator output in the Dymola environment via using physical components from the TRANSFORM (Transient Simulation Framework of Reconfigurable Models) library. Its nuclear steam supply system (NSSS) was an iteration of the Generic i-PWR class of TRANSFORM library, and its neutronic core model was based on the point reactor kinetics equation (Equation 1) with six neutron precursors groups (Equation 2).

$$\frac{dn(t)}{\mathrm{dt}} = \frac{\rho(\mathrm{t}) - \beta}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i \tag{1}$$

$$\frac{dC_i\,(t)}{\mathrm{dt}} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i\,(t) \tag{2}$$

Where, n(t) is the neutron density, t is the time, ρ(t) is reactivity, β is the total fraction of delayed neutrons, β_i is the fraction of delayed neutrons in i-th groups, Λ is the prompt generation time, λ_i is the decay constant, C_i is the precursors' concentration. Furthermore, the control systems were devised in the Simulink environment and used the ideal Stodola's ellipse law turbine. Figure 3 illustrates the diagram of this simulator. [12]

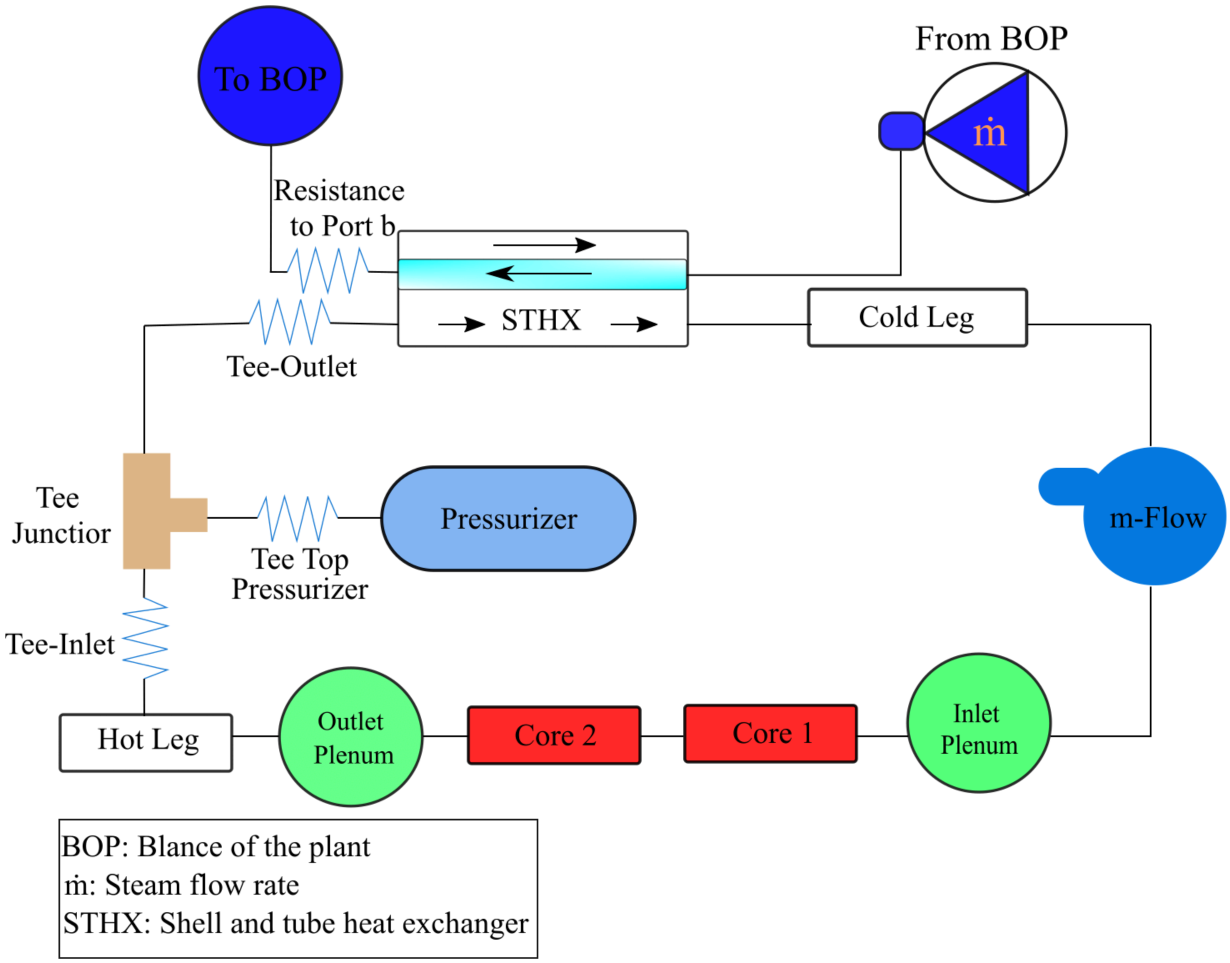

Figure 3. The flow chart of the primary loop developed by the Oak National Laboratory. Modified from [12]

For the Idaho National Laboratory (INL) team, they created a nuclear reactor subsystem model for the Nuclear-Renewable Hybrid Energy System project. They focused on the primary heat transfer loop including the reactor core, Pressurizer, and steam generator. The elements of the reactor core contained the reactor dynamics model, fuel heat conduction model, and flow channel model. All of the reactor cord models were applied to the point reactor kinetics equation (Equations 1 and 2) and 2-D conduction equation. The pressurizer was a nonlinear and nonequilibrium model for managing the steam pressure and water level that uses a first-order transfer function with empirical time constants and was calibrated or regulated by spray and heater systems. The design of the steam generator was one through helical-coil tube accommodating eight modules of steam generators, and the total heat transfer area was 1150m^2. Figure 4 shows the diagram of the primary heat transport loop. [17] Overall, ORNL and INL collaborate on different aspects of simulations. Both require multiple team developers to adhere to interface definitions to achieve model compatibility.

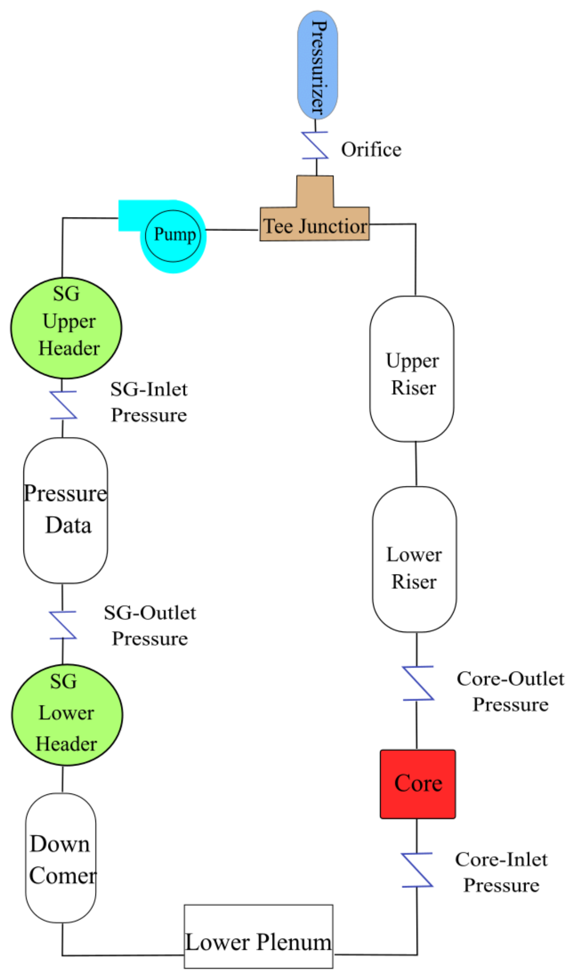

Figure 4. The scheme of the primary heat transport loop developed by the Idaho National Laboratory. Modified from [13]

2.1.2 Fortran Based Model

The term Fortran is a compiled programming language compared to the previous mentioning Modelica code, which is an object-oriented modeling code. INL and North Carolina State University developed the i-PWR simulator based on Fortran code. This i-PWR simulator's thermal power was 500 MWth, and generator output was 158 MWe deriving from the NuScale power reactor and existing IRIS (PWR) simulator with 1000MWth and 335 MWe. This simulator contained a pressurizer, a reactor core using the point reactor kinetic equation (Equations 1 and 2), a decay heat model, a hot channel model, a primary loop flow, a steam generator, and a balance of plant components. In addition, it can operate the i-PWR into forced or natural circulations. In INL and North Carlina State University's research, they not only invented the simulator but also analyzed the simulator performances. They operated 100% turbine load and down to 90% to observe the transient including primary pressure, average coolant temperature, feed flow rate, and normalized power. [14]

The IEEE members and Bikash Poudel et.al [15] presented and developed the i-PWR type simulator in Siemens PTI PSS/E platform with Fortran code integrating the IEEE standard GGOV1 turbine-governor system approximating the internal source prior to turbine valve with a first-order transfer time function. In this function, they considered time as a constant function. Figure 5 shows the framework of the IEEE simulator. This simulator's generator output was 45 MWe according to the NuScale design. For the reactor core, they applied the point reactor kinetic equation (Equations 1 and 2), and its main coolant mass flow was in reference to the natural circulation described as a function of thermal power. Besides, the steam generator was simplified into three lump models, which was different from the common helical model design. They analyzed the steady-state and dynamic performance of i-PWR simulations. For example, they adjusted the reactivity via moving control rods, valve opening, and power load for the dynamic study in 10 minutes. [15]

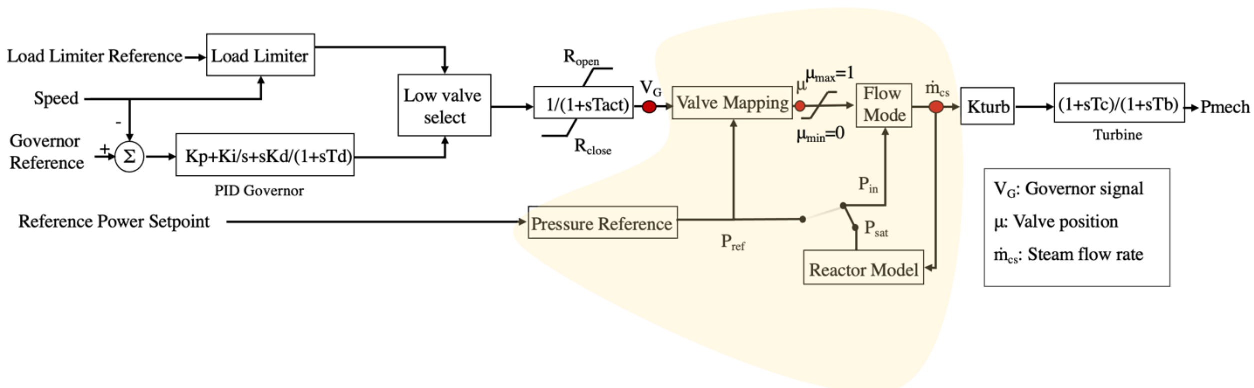

Figure 5. The outline of the IEEE simulator model based on Siemens PTI PSS/E. Modified from [15]

2.1.3 PC-Based PCTRAN MODEL

The micro-simulation Technology used Personal Computer Transient Analyzer (PCTRAN) software to develop China ACP 100, NuScale SMR, and Korean SMART simulators. Figure 6 presents the NuScale SMR simulator's interface, reactor core, and auxiliary system. All of those reactors' characteristics have been illustrated in table 1. In Korea, Juyoul Kim et.al [16] used this simulator to investigate the transient of SMART. Besides, they conducted loss of coolant accidents and normal operating situations to comprehend SMART reactor performances including pressure, temperature, thermal power, flow mass rate, and design features. In E. Rafee et.al [17] research, they compared 6 different types of reactors based on PCTRAN simulators including China ACP 100 and NuScale SMR within the MATLAB-Simulink environment during the loss of coolant accidents. They also considered fission product poisons such as Xenon-135 and Samarium-149 impact on reactivity. Furthermore, they evaluated the heat transfer via monitoring the variation of pressurizer level and steam generator wide range.

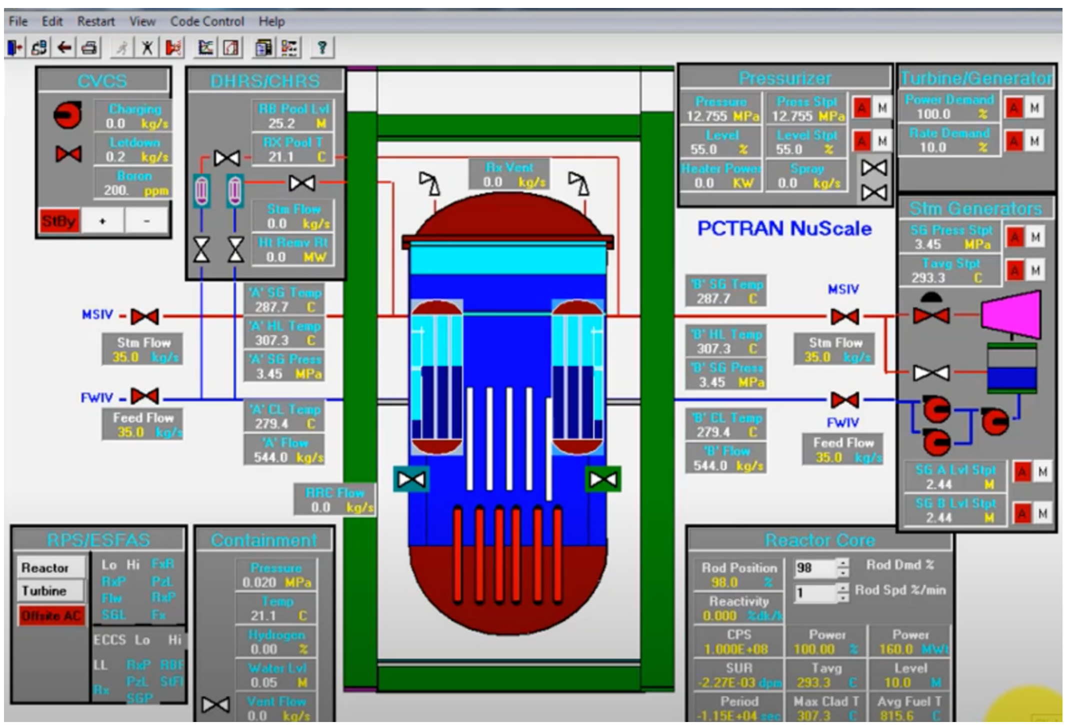

Figure 6. The interface of NuScale SMR PCTRAN simulator developed by micro-simulation Technology.

2.2. Description of the IAEA Model

The IAEA i-PWR model simulator was developed by Tecnatom in 2017. This company supplied and assisted Norway to develop its SMR simulator in 2022. [18] The objective of this simulator is to provide the i-PWR reactor's operational characteristics and virtual practice opportunities for reactor operation. With this purpose of the IAEA i-PWR simulator, the simulator allows the operators to realize the reactor transient, perturbation, response behaviors, and interactions of the complex system. IAEA allows member states to apply and process this simulator. It makes more international countries can involve in the study of the i-PWR reactor and cooperate with each other. Therefore, owing to the properties of this simulator, this research selects the IAEA simulator to investigate the fluctuations and trends of normalized power and the generator load during daily operations.

This IAEA simulator contains a reactor core, reactor coolant system (RCS), main steam system (MSS), feedwater system (FWS), Turbine system (TUR), generator system (GEN), condenser system (CNR), protection and control system (PCS), and safety system including automatic depressurization system (ADS), gravity-driven water injection system (GIS), pressure injection system (PIS), and passive decay heat removal system (PDHR). Figure 7 shows the overall system configurations. The scope of this simulator concentrates on those main systems, neglecting the other auxiliary or supporting systems such as shutdown cooling system, containment vent, turbine auxiliaries, off-site power, etc. In addition, owing to educational purposes, this simulator does not handle complicated thermal hydraulics. Besides, this reactor can operate under natural circulation (NC) and forced circulation (FC) depending on demands. Moreover, there are several initializations (ICs) that can be selected when loading this simulator. For instance, IC1 runs the reactor at 100% normalized power at the beginning of life (BOL) and natural circulation (NC). IC5 executes the reactor at 100% power at the middle of life (MOL) and forced circulation (FC). IC12 conducts the reactor at 0% power, forced circulation (FC), and hot reactor critical condition Keff=1. [19]

In general, this model can demonstrate the reactor transient and balance of plant behaviors during normal operation. Furthermore, this model has malfunction options involving valve fail opening, pipe break, loss of coolant accidents, station blackout, etc. Hence, the reactors' status alterations can be investigated while those situations occur. [19, 20]

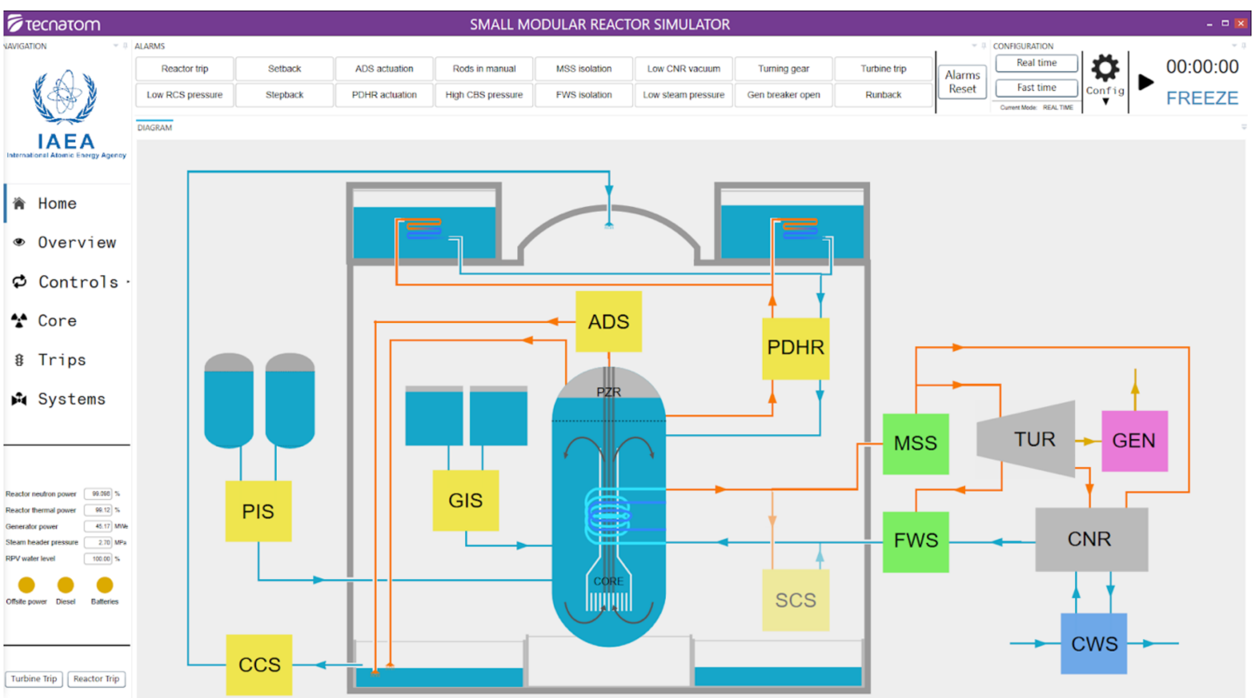

Figure 7. The overview scope of the IAEA simulator

2.2.1 Reactor Core and Reactor Coolant System (RCS)

The build-in fuel is UO_2 with 4.95% enrichment which can produce 150 MWth when running full power of the reactor. This model has 24 assemblies, and each assembly has 17*17 fuel pins. The fuel pins' length is approximately 1.35 meters, and the equivalent diameter is about 1.2 meters. To decrease the leakage and contamination of fission products, this model has two barriers. One is the clad, the other is the reactor construction. Besides, the reactor core, steam generator, and pressurizer are accommodated in the stainless-steel reactor pressure vessel. The volume of the reactor pressure vessel and pressurizer is 80.78 m^3, and 8.078 m^3 respectively. The reactor's saturation temperature is 344.8°C at 15.5 MPa. Moreover, the steam generator consists of two helical tubes and they can serve as the barrier for the primary and secondary sides' steam of the reactor. Each helical tube contains 506 tubes and its thickness is 0.9 millimeters (mm). The diameter of the tube is 16 mm. The standard coolant flow is 424 Kg/s passing through the reactor pressure vessel, and if it is under forced circulation, the four horizontal pumps will draw the coolant flow above the reactor core which can be observed in Figure 7.

The total reactor control rods consist of a control group and a shutdown group. The control group is used to normally adjust the reactor's reactivity. Whereas the shutdown group is utilized for emergency cessation of the reactor during the loss of power or scram events. When urgent accidents happen, the control rods of the shutdown group will automatically drop. The boron concentration also has the ability to affect reactor reactivity. The pressurizer, which integrates into the reactor pressure vessel, maintains the coolant pressure at 15.5MPa during operation by balancing the steam and water. If steam is over the equilibrium limitation (17.05MPa), the relief valve will open to depressurize. Moreover, the reactor safety system such as PIS, ADS, and GIS will be active to keep the reactor in a safe condition. Figure 8 displays the reactor core overview including the fuel temperature, neutron flux, boric acid concentration, and reactivity of the control rod, fuel, moderator, boron, and so on. [19]

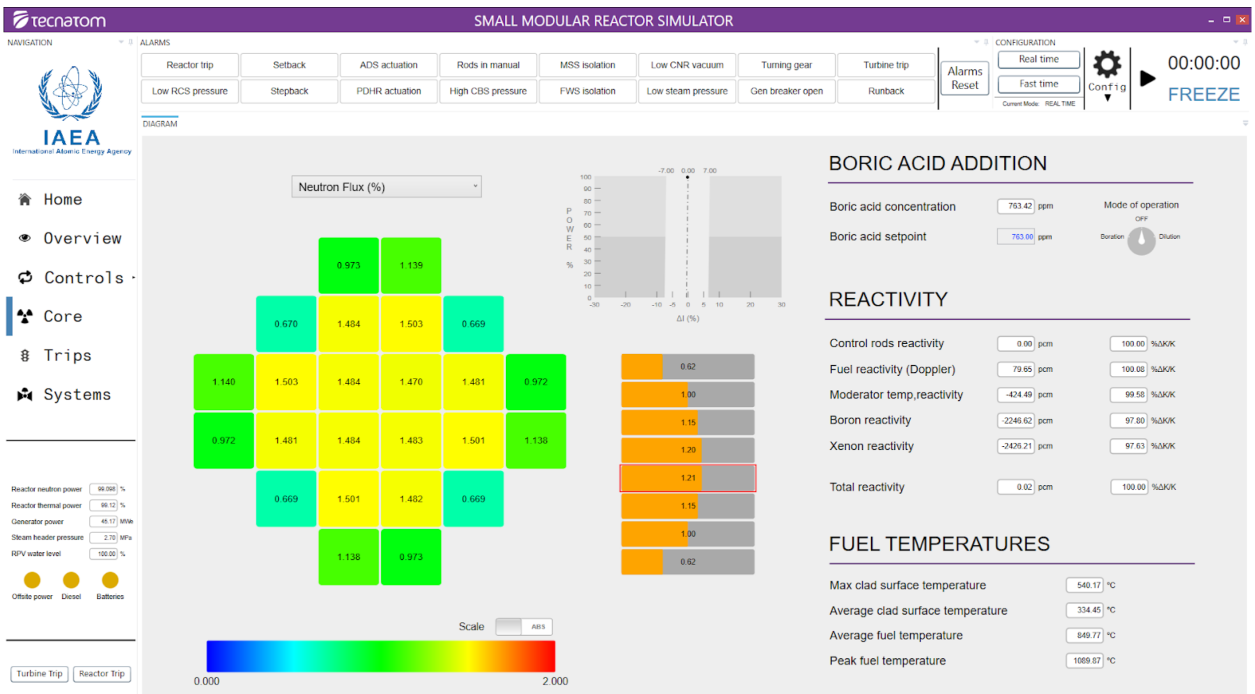

Figure 8. The reactor core overview of the IAEA model

2.2.2 Main Steam System (MSS)

The main steam source comes from the steam generator. MSS will predominantly drive the steam to the turbine generator or to the condenser system. The diameter of the main steam line and the bypass line are 400 mm and 300mm separately. The bypass line connects to the condenser system (CNR) passing through the MSSV09 (control valve) and MSSV10 (isolation valve). The main steam line drives steam to the turbine system via commuting to the MSSV07 (control valve) and MSSV08 (isolation valve). Regularly, the MSS flow stays 77 Kg/s at 2.7MPa during full-power operation. MSSV01 to MSSV03 are relief and safety valves that protect the reactor far from pressure exceedances. The isolation valves can break the steam to the turbine to regulate the flow during the turbine trip. The top level of the diagram in Figure 9 presents the valves and pipeline for MSS. [19]

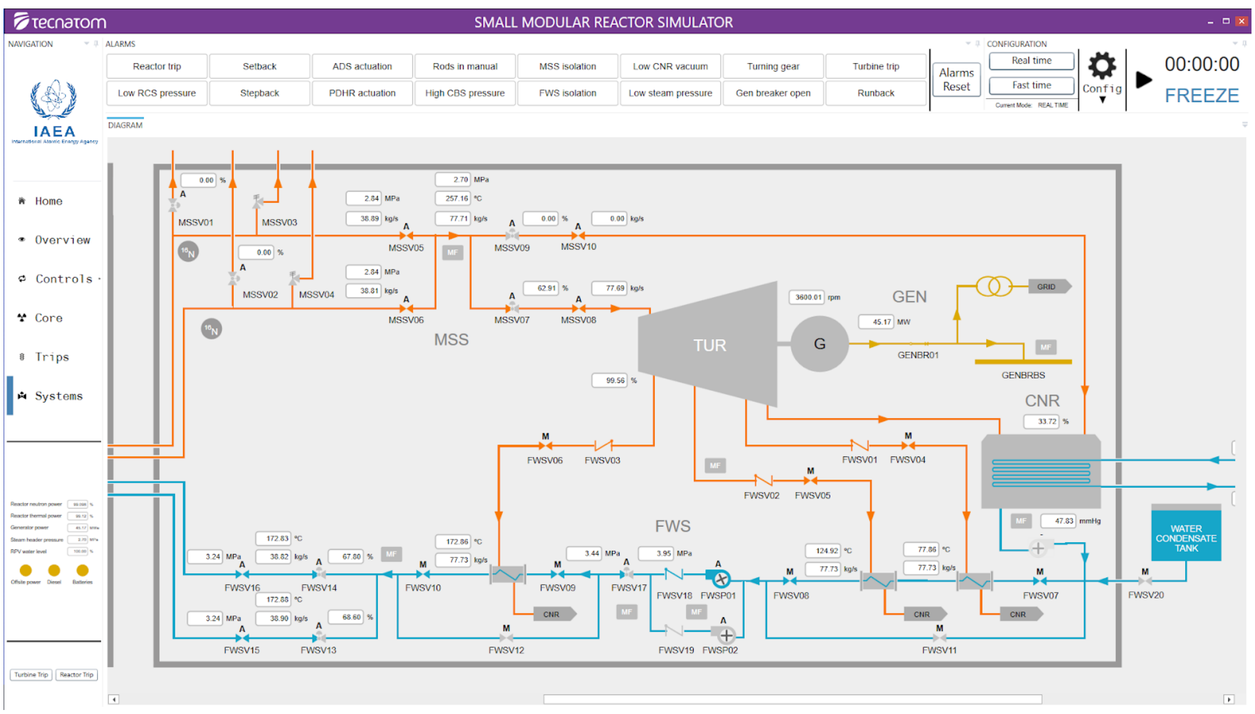

Figure 9. The pipeline diagram contains MSS (upper part), FWS (bottom part), condenser system (CNR), and TUR (middle).

2.2.3 Feedwater System

Figure 9 shows the FWS component in the bottom part. There are two sources of water. One comes from a water condensate tank. The other is from the turbine and some of it has already been processed from the CNR which works under a vacuum (47.83 mmHg). The water from the condenser has to go across three heaters to preheat to approximately 173℃, and pump (FWSP01) through the reactor core system becoming steam. The number of heat exchangers has to go through depending on the water temperature. FWSV14 and FWSV13 are control valves letting water into RCS. If heaters have some problems, the heating bypass valves (FWSV12 and FWSV11) will work to decline the influence. The water condensate tank only can supply two hours in the event of feedwater or main stem loss. The diameter of the feedwater pipeline is 150 mm. This simulator tube's diameter is considerably smaller than the PWR type mentioned in section 1, Figure 2. [19]

2.2.4 Turbine and Generator System

The turbine is a mechanical rotor that is rotated by steam, and the steam is generated from the reactor. The turbine requires a frequency of 60 HZ and a speed of 3600 rpm to keep the turbine in a safe condition. The generator is driven by the turbine which produces electricity. Its maximum guaranteed output is 45MWe with three-phase AC. In the real, the generator consists of a shell, stator, bearings, excitation, and generator auxiliary system. Contrastingly, this simulator only focuses on the shell, stator, and bearings. Excitation is the electromagnetic induction process that leads mechanical energy to convert to electricity via passing through a magnetic field. Figure 9 depicts the outline of turbine and generator systems. [19]

2.2.5 Circulating Water System (CWS)

The circulating water system has two modes. One is an open-loop connecting to a lake or sea. The other is a closed-loop coupling of a cooling tower which can be observed in Figure 10. The purpose of the circulating water system is to pump adequate water to supply the condenser to condense steam and transmit temperature away from the condenser. Furthermore, it can help the condenser maintain the vacuum. [19]

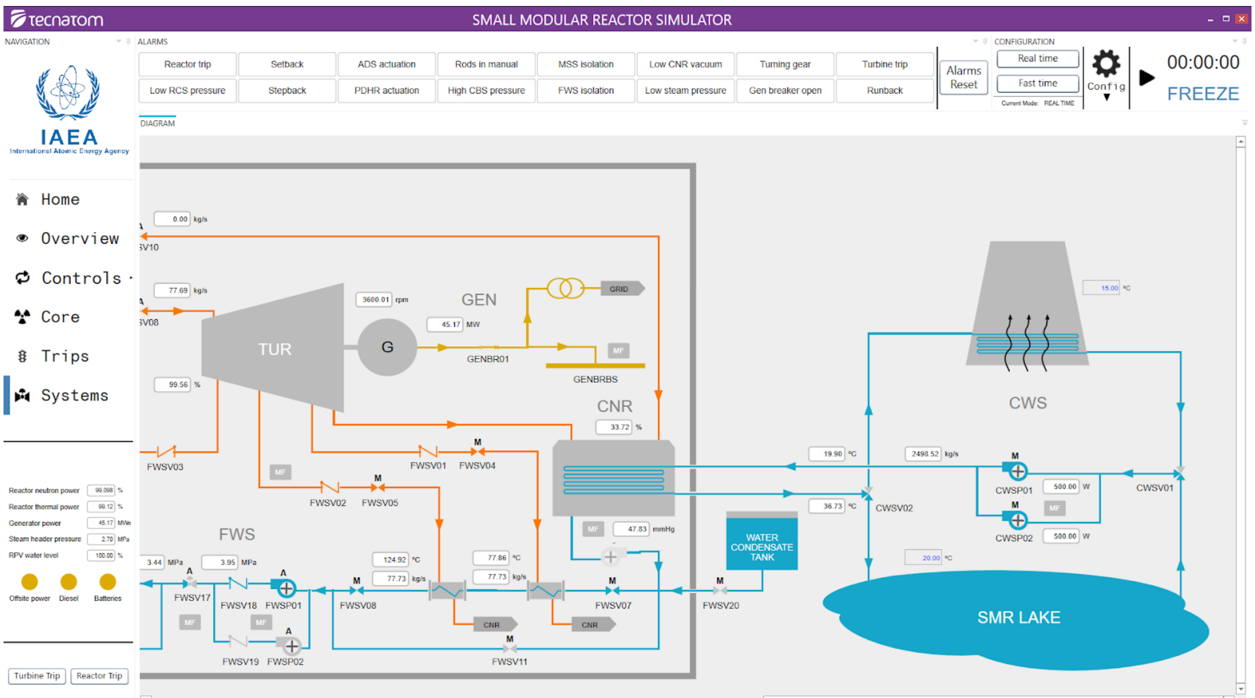

Figure 10. The pipeline schematic layout of the Circulating Water System (CWS).

2.2.6 Passive Safety System

The Passive Safety System involves an automatic depressurization system (ADS), Pressure Injection System (PIS), Gravity Injection System (GIS), and Passive Heat Remove System (PDHR) shown in Figure 11. ADS is controlled by three valves (ADSV01, ADSV02, and

ADSV03), whose diameter is 101.6mm. The main function of ADS suppresses the pressure and discharges superfluous steam to the suppression pool. This function can prevent the reactor core from melting and create more time to deal with emergency events such as LOCAs. For PIS and GIS, they will release borated water (2500ppm), when the reactor vessel's pressure is below 5MPa. Therefore, when the pressure of the reactor vessel is lower than 5 MPa by ADS active, PIS and GIS will operate by injecting borated water into the reactor. In this simulator, PIS and GIS have two tanks with borated water and two control valves. GIS has two control valves (GISV02 and GISV04); PIS control valves are PISV02 and PISV03. The function of PDHR removes the heat from the core. Commonly, PDHR will be active due to isolations of FWS and MSS systems losing the cooling ability. Thus, PDHR's pool will condensate the steam to reduce the temperature and pressure. Its pool can maintain a minimum of seven days of natural circulation with a volume of 148.5 m^3 in this simulator. [19]

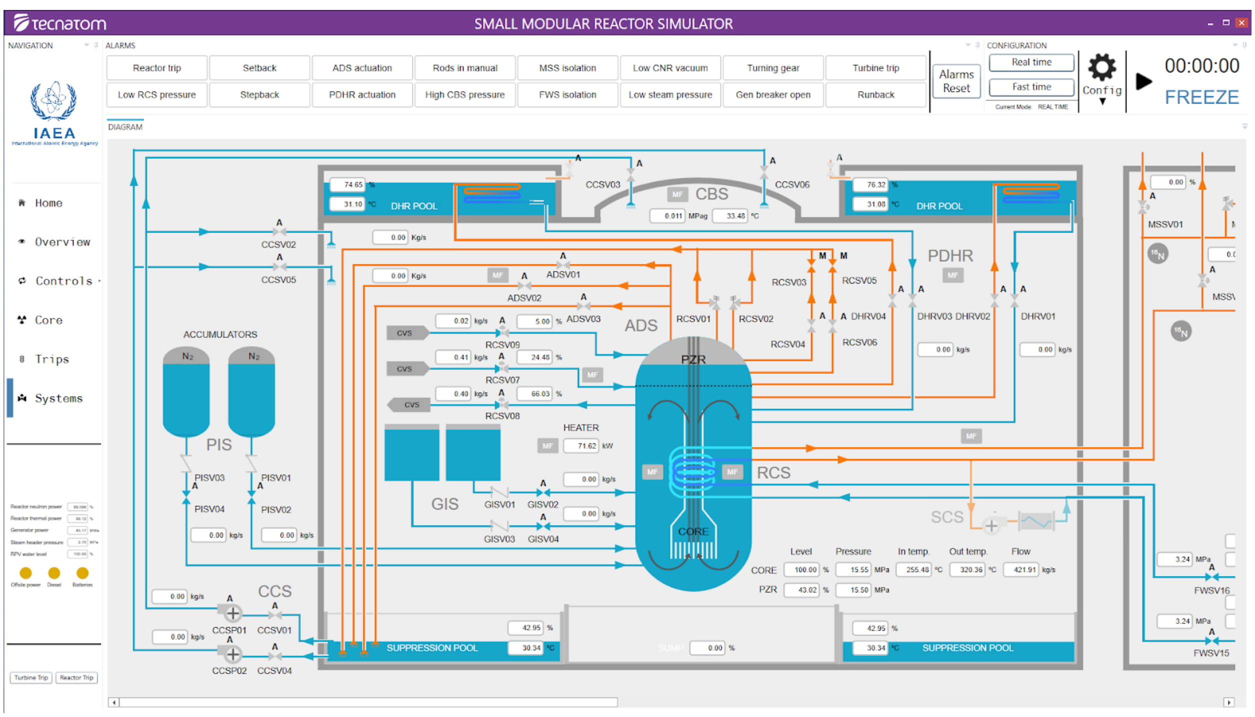

Figure 11. The overview of the reactor vessel with the passive safety system

2.2.7 Protection and Control System (PCS)

The main role of the Protection and Control System (PCS) is to make sure the reactor is secure. If there are some problems, it will show the alarm or trips to let the user know and send the signal to the safety system. For the control system, it maintains the balance of the plant during the operation. The control panel includes pressurizer water level, pressurizer pressure, feedwater, turbine, and rods position. Each panel can modify each parameter. For example, Figure 12 displays rods position control. The operator is able to switch the plant mode to determine the loading. To be more specific, if the model is on the reactor loading, the operator can adjust the setpoints of the reactor power demand and reactor power rate. In addition, the user is capable of moving the control rods according to executing cases and performances. In contrast, if the user selects turbine loading mode, the reactor will follow the turbine load demand and turbine load rate setpoints. In order to achieve those actions and outcomes, the reactor will automatically adapt the valve opening range, steam, and flow rate. Moreover, this simulator allows users to modify some values such as valve opening range, pressurizer water level, pumps active, etc. Figure 13 demonstrates the turbine control panel and manual actions. [19]

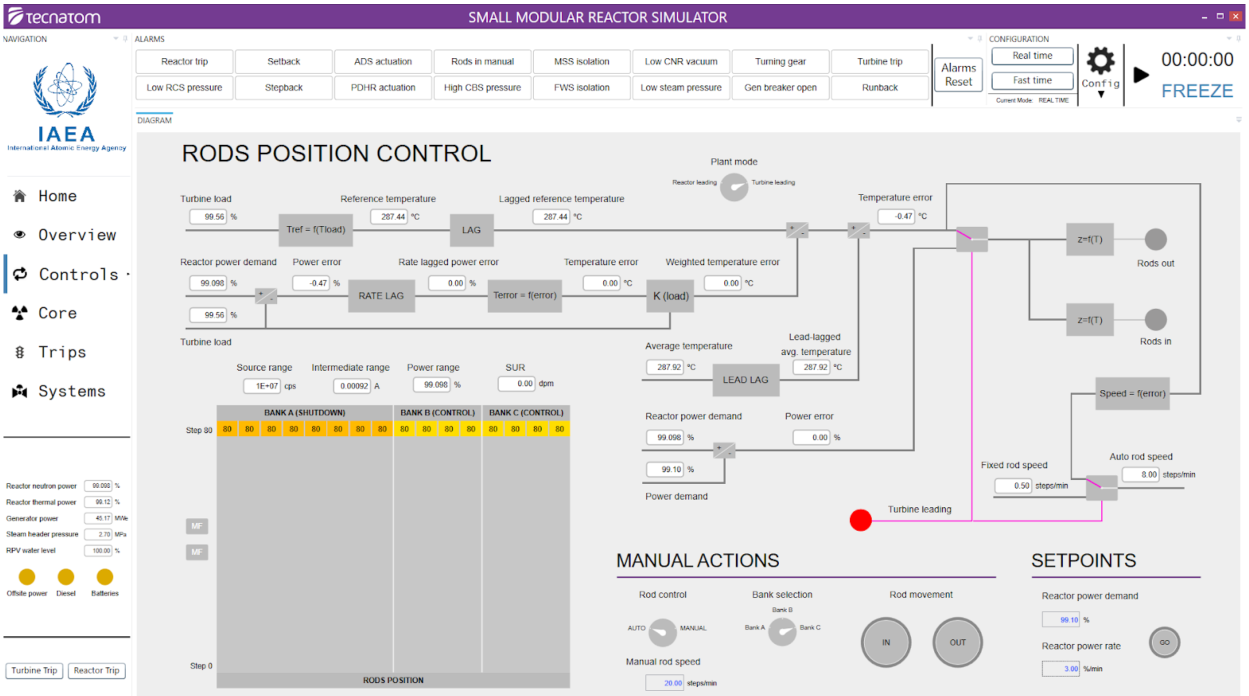

Figure 12. The rods position's control panel of the control system.

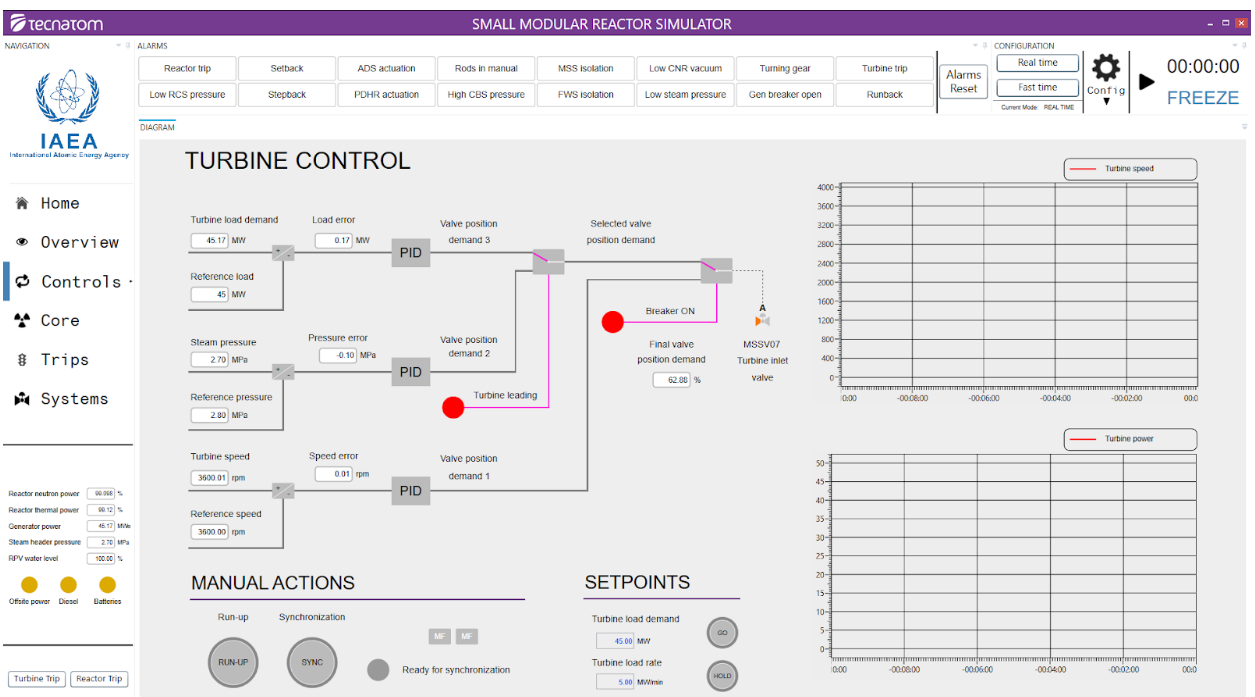

Figure 13. The turbine control platform of the control system

2.2.8 The Description of how conducting the IAEA Simulator for This Study

This research uses the default of initialization (IC#1). The default settings are running 100% power at the beginning of life (BOL) and natural circulation (NC) for inherent safety. In order to achieve the scenario setups, this study chooses the plant mode in the rod position control interface pages as shown in Figure 12, and adjusts the reactor power setpoint to meet the requirement of the scenarios. Once the reactor thermal power attains the goal, saving as new initialization conditions. Then, reload the new initializations in the configuration menu to ensure the reactor status is a clean core without fission products or neutron poisoning accumulations. Besides, the reactor power rate will be set at 3% per minute to comply with the regulation as the steady-state scenarios settings. The reactor power rate will be placed between 3~5% to achieve dynamic scenarios of daily operation demands

Once the reactor simulation time reaches the settings, clicking the freeze button and exporting output data to excel that option can be found in the configuration interface. The output data includes several parameters such as pressurizer water level, total flow rate, control rod worth, average coolant temperature, core reactivity, generator load, pump power, thermal power, normalized power, and so on. Furthermore, during the simulation time, this research ensures that no reactor trips happen to mimic the real regular operating situations.

2.3 Summary of the i-PWR simulators features and capabilities

Table 2 illustrates the outline and differences between the IAEA and previous i-PWR reactors' simulators. It can be noticed that the simulator designs of reactor thermal power usually are 150 or 160 MWth and the generator load is between 45 to 158 MWe. In addition, the neutronic model of the simulators is established on the point kinetic reactor equation with six neutron precursors (Eq1 and Eq2). However, the thermal-hydraulic model usually depends on the developer. According to the present i-PWR reactors' design, steam generators commonly are helical types. Except for the IEEE's simulator, the other simulators' steam generator type is the same as the current real-world design. Furthermore, the PCTRAN and the IAEA simulators can not be coupled with a hybrid energy system. Even though they only simulate the nuclear power plant situation, which is unlike other models having the ability to annex renewable energy systems, including solar power plants, wind power plants, hydraulic power plants, etc, the IAEA simulator still has numerous advantages. For instance, the IAEA simulator is open source, that provides to the IAEA member states to use. It does not reflect any specific vendor's design and is easy to run on Windows computers. Therefore, this research selects the IAEA simulator as a tool to analyze the reactor transient performance and trends.

Table 2 The overview and comparison of i-PWR simulators

Simulator	IAEA	PCTRAN-NuScale	Modelica code-based model	Fortran based model	
Open source	O	X	X	X	X
Max reactor power (MWth)	150	160	160	500	150
Max generator load (MWe)	45	60	53	158	45
Neutronic model	NEMO[a]	Point Reactor Kinetic equation with six precursors			
Thermal hydraulic model	TRAC_RT[b]	Lumped-loop approach with two-phase critical flow	2-D conduction and conservation law		Hann's model
Steam Generator model	Helical	Helical	Helical	Helical	Three lumps
Coupling hybrid energy system	X	X	O	O	N/A
Environment platform	Windows	Windows	Matlab simulink	Matlab simulink	Siemens PTI PSS/E
Based design model	N/A	NuScale	N/A	IRIS, NuScale, SMART	NuScale
Developer	Tecnatom	Micro-simulation Technology	ORNL, INL, NC state university	ORNL, INL	IEEE
Reference	IAEA [19]	Micro-simulation Technology [21, 22] & NuScale [4]	Richard Bisson et.al [12]& Mikkelson, D. et al [13]	Bragg-Sitton et.al [14]	Poudel, B. et.al [15]

a. The detail of NEMO will be discussed in section 3.1

b. The description of TRAC_RT will be discussed in section 3.2

3. TRANSIENT BEHAVIORS AND SCENARIOS

3.1. Reactor Core

The IAEA i-PWR simulator utilizes a 3-D neutronic model based on a modified one-group diffusion theory with nodal feedback and couples with thermal-hydraulics code TRAC-RT, which was developed by TECNATOM company. This simulator uses Equation 3 to figure out the space and time-dependent distribution of the power production in this neutronic model (NEMO). Where the term $P(\vec{r},t)$ is power depending on the time and space, $T(z,t)$ is the 1-D amplitude function with time and Z-axis direction-dependent, $S(\vec{r},t)$ is the 3-D normalized shape function relying on the time and location.

$$P(\vec{r},t) = T(z,t) \times S(\vec{r},t) \tag{3}$$

To calculate the P more efficiently and solve the 3-D shape function at a low frequency, the derivation of P can be articulated as a mesh region integral quantity (P_L). Equation 4 demonstrates the simplified differential equation for P_L with delayed neutron sources (Equation 5). Where, Λ is prompt neutron generation time, P_L is the energy production rate (power) in mesh region L, β is the total fraction of delayed neutrons, β_i is the fraction of delayed neutron in ith groups of precursors, $\rho_{\infty L}$ is the local reactivity in mesh region L ignoring leakage, W_{ML} is a coupling coefficient, P_M is the net neutron current from node M to node L, l_L^H is the probability that neutrons produced in region L escaping from the core horizontally, l_L^V is the probability that neutrons produced in region L, escaping from the core in the vertical direction, S_L^B is the production rate of neutrons from no-fission sources in mesh region L, C_{iL} is the delayed neutron production rate from precursor group i in mesh region L, λ_i is the decay constant of the delayed neutron precursor in i group. [19]

$$\wedge \frac{dP_L}{dt} = (\rho_{\infty L} - \beta)P_L + \sum_{M \neq L} W_{ML}P_M - \sum_{M \neq L} W_{LM}P_L - l_L^H P_L - l_L^V P_L + \sum_i C_{iL} + S_L^B \quad (4)$$

$$\frac{dC_{iL}}{dt} = \lambda_i(\beta_i P_L - C_{iL}) \quad (5)$$

$$\wedge \left(\frac{dS_L}{dt} + S_L \frac{d\, ln(T_k)}{dt}\right) = (\rho_{\infty L} - l_L^H - l_L^V - \beta)S_L + \sum_{M \neq L}\left(\frac{W_{ML}T_{\dot{k}}}{T_k} - W_{LM}S_L\right) + \frac{\sum_i C_{iL} + S_L^B}{T_k} \quad (6)$$

Equations 3, 4, and 5 can be rewritten into Equation 6. In Equation 6, the function of the 3-D shape can be calculated based on the assumption of $\frac{d\, ln(T_k)}{dt}$ which can be derived from the two most recent1-D kinetic time steps, and T_k is the term of amplitude value in the axial layer K from the Equation 3 of 1-D amplitude function. After calculating and defining the space-time equation, it should be perceived that in 1-D kinetics and 3-D shape calculation results are not normalized, so to fix this issue, it has to renormalize S_L. Generally, the 1-D kinetic model is based on the eleven groups of decay heat formulations, and the delayed neutron sources use six groups of precursors and integrate those data and time for 3-D calculations. Besides, the IAEA simulator's core behaviors comply with conservative laws such as the balance of mass, momentum, and energy. Furthermore, the cross-section is presented implicitly in the reactivity and coupling coefficients, which is not a constant function. Moreover, the reactivity calculation in this model is depicted in Equation 7, considering the fuel, moderator, control rods, xenon, boron, samarium, and void reactivities.

$$\rho_{\infty L} = \rho^0_{\infty L} + \rho^{DOP}_{\infty J} + \rho^{MOD}_{\infty J} + \rho^{BOR}_{\infty J} + \rho^{XE}_{\infty J} + \rho^{SM}_{\infty J} + \rho^{RCC}_{\infty J} + \rho^{VOID}_{\infty J} \quad (7)$$

Where, L is a neutronic node, J is a thermal-hydraulic node in the L node region, $\rho_{\infty L}$ is the node reactivity-infinity, $\rho^0_{\infty L}$ is the initial condition of ρ_∞ based on the local burn-up and the fuel type, $\rho^{DOP}_{\infty J}$ is the doppler effect contribution of the reactivity, $\rho^{MOD}_{\infty J}$ is the moderator node reactivity, $\rho^{BOR}_{\infty J}$ is the boron node reactivity, $\rho^{XE}_{\infty J}$ is the xenon node reactivity, $\rho^{SM}_{\infty J}$ is the samarium node reactivity, $\rho^{RCC}_{\infty J}$ is the control rods node reactivity, $\rho^{VOID}_{\infty J}$ is the node reactivity of void. [19]

3.2. Description of TRAC_RT Code

The TRAC_RT (Transient Reactor Analysis Code - Real time) code is utilized to develop and compute the thermal-hydraulic behaviors of the reactor core, coolant (RCS), main steam (MSS), feedwater (FWS), turbine (TUR), and containment systems. The characteristics of the TRAC_RT code are using the non-homogeneous, non-equilibrium two-fluid model, the 1-D heat conduction model, the 3-D vessel model, the boron model, and the non-condensable model. The basic solution of thermal-hydraulics in this simulator is via the continuity equation, the conservation of energy equation, and the conservation of momentum equation. Table 3 shows the overview of this simulator’s thermal-hydraulic parameters when loading full power inputs to the TRAC_RT code. [19]

Table 3 The summary of the IAEA i-PWR simulator’s thermal hydraulics parameters. Modified from [19]

Thermal hydraulic parameters	Value	Units
Reactor neutron power	100	%
Thermal power	150	MW

Table 3 Continued

Generator power	45	MWe
Primary pressure	15.5	MPa
Pressurizer level	43	%
Core level	100	%
Core flow	422	Kg/s
Inlet core temperature	255.51	°C
Outlet core temperature	320.36	°C
Average core temperature	288	°C
Steam header pressure	2.72	MPa
Steam reheating	29	°C
Steam flow	77	Kg/s
Feedwater flow	77	Kg/s
Feedwater temperature	173	°C
Turbine speed	3600	rpm
Condenser vacuum	47.81	mmHg
Containment pressure	0.011	MPa
Containment temperature	33.4	°C

3.3. Description of Electrical Network

There are several methods to solve and calculate the voltage and intensity values such as iterative methods, algebraic methods, and so on. This simulator uses algebraic methods to compute the solution via converting the data into a matrix to determine the electrical variables (Equation 8). It also applies Ohm's law (Equation 8), Kirchoff laws, and the equivalent Norton circuit. Where [I] is the matrix of electrical intensity, [Y] is the matrix of admittances, and [V] is the matrix of voltage. [19]

$$[I] = [Y] \times [V] \quad (8)$$

3.4. Transient Behavior

When the reactor's thermal power decrease via inserting control rods, the fuel temperature declines, which leads to main steam flow (MSSFT01_TR and MSSFT02_TR), total flow water rate, coolant temperature, and PZR (pressurizer) level becoming lower. Meanwhile, the Generator load will start to drop, and turbine control valves (MSSV07) begin to close. Contrastingly, if the reactor thermal power increases, the fuel temperature starts to rise, causing steam rate, total flow rate, and PZR level increment. The generator load and turbine valves' opening range begin to rise consequently. However, except for accidents, the pressurizer pressure keeps 15MPa no matter the change in thermal power and generator load. [19, 20]

To better comprehend the details of the i-PWR reactor's transient performances, this research uses plant mode and turbine mode to operate and control the IAEA simulator to shut down and startup the reactor as an example. The result shows in Figure 14. It can be observed that the normalized power stays close to 100% in the first one-half hour. During the reactor shutdown process, it notices that the generator load, average fuel temperature, total flow rate, and PZR level become decreasing in Figure 14. When the generator load is from 45 MW to 6.75MW, Figure 14 presents the fluctuation of normalized power, average fuel temperature, and PZR level due to inserting control rods. When the operating reactor goes down to 0 MW, meaning completely shut down, the

FWS flow total rate has larger oscillations, and the PZR level keeps fluctuating by the reason of neutron poisoning and decay heat. Approximately, in the complete shut-down (Normalized Power and Generator load are zero) interval of 2 hours 45mins to 4 hours 45 mins, Figure 14 demonstrates the PZR level, and total flow rate still has some variabilities. After complete shutdown, the turbine speed from 3600 revolutions per minute (rpm) drops to 5 rpm. After 4 hours 45 mins, the reactor startup and normalized power responses faster than the generator load because producing enough steam to push turbine speed back to 3600 rpm takes some time. Besides, the boron concentration decreases. Furthermore, the fuel temperature, total flow rate, and PZR level react immediately. The total flow rate has a high frequency of fluctuations compared to other parameters when the reactor startup. After the reactor's generator load achieves 20 MW, the total flow rate's fluctuation decrease. However, the PZR level still oscillates until the reactor is back to full power. The average fuel temperature fluctuations are similar to normalized power from beginning to end. Figure 14 indicates when operating the reactor goes back to the set point of 100% power, the parameters of the reactor, such as normalized power, average fuel temperature, PZR level, and so on are unstable and have more oscillations, compared to the initial. Overall, When the reactor starts to shut down, the PZR level, normalized power, and average fuel temperature begin oscillating, and the total flow rate fluctuates lately. During the complete shut-down interval, the total flow rate and PZR level still have oscillations. Although average fuel temperature still has few fluctuations, it is relatively stable, compared to the total flow rate and PZR level.

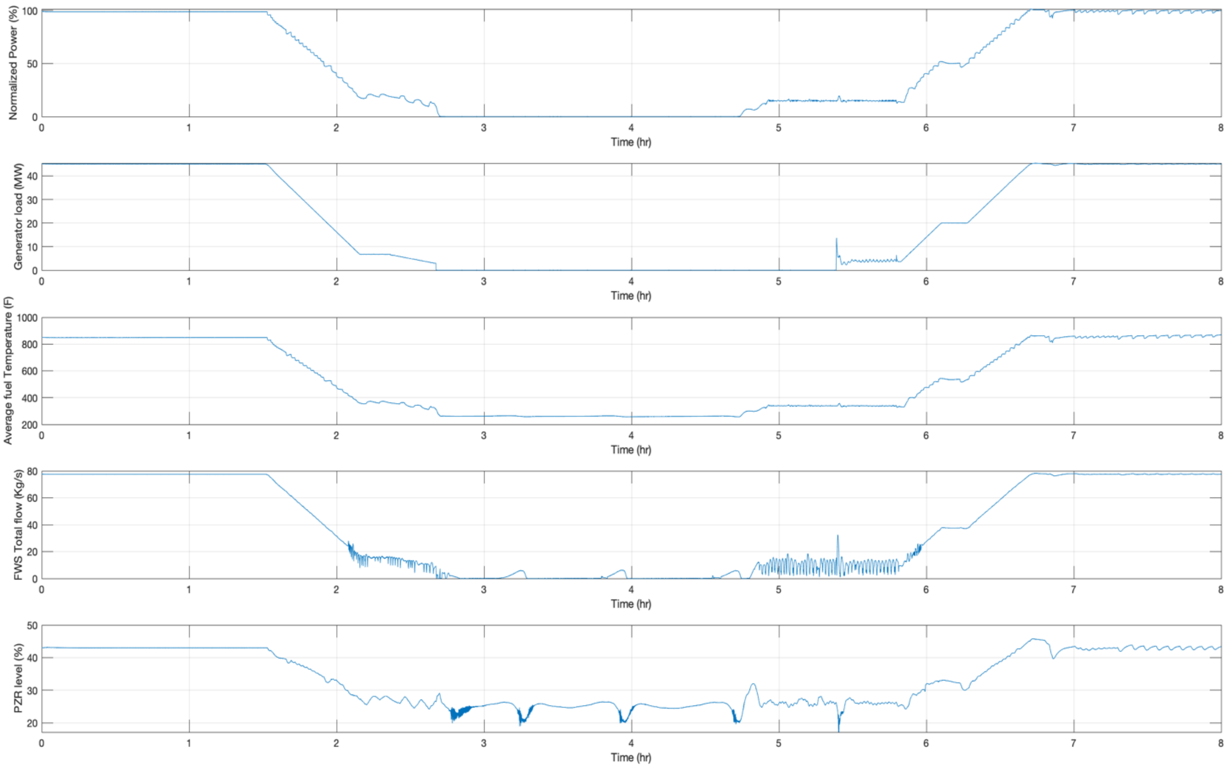

Figure 14. Graphics from top to bottom are normalized power, generator load, average fuel temperature, total flow rate, and pressurizer level (water level in the reactor vessel) versus time during the reactor transient from shutdown to startup.

3.5. Scenarios Setup

This research separates the scenarios into two categories. One is base-load for investigating the reactor steady-state transient behaviors. The other is load-following for prospecting the reactor dynamic transient performances during nominal daily operation. However, there are no daily operation guidelines for i-PWR reactors, so this study utilizes the common operation instructions from conventional reactors, especially for PWRs. In addition, different brand reactors have distinct loading regulations owing to different fuel types and burn-up ratios. Hence, it is essential to refer to the different countries' regulations and recommendations.

The term base-load means operating reactors at steady full load power, and only under two conditions, that operators will reduce power or shut down the reactor. The conditions are planned refueling or periodic maintenance, and unplanned situations such as emergency accidents. Base-load is the nuclear power plant driven, making the reactor easier to operate and more efficient in utilizing the nuclear fuel. Besides, there are some advantages of the base-load mode. For instance, operating reactors with a constant load mean less complicated and better to predict the reactor performance, it makes the design and licensing of plants simpler, and compared to non-baseload mode, it does not need more times of maintenance and monitors. Furthermore, when reactors operate at full load, usually it is the lowest cost for running nuclear units. [23, 24] Nevertheless, the load-following mode in some ways can decrease the cost by reducing over-generating electricity. In A. Lokhov's report [25], it points out that the United States basically operates the reactor in base-load mode on account of the US Code of Federal Regulation (10 CFR Part 50 and Part 55). It mentions the alternation of reactor power or reactivity should be consented to by knowledgeable or senior licensed operators. Based on this reason, it limits the manipulation for operators adjusting the reactor power.

Load-following is an electricity grid driven to generate just the right amount of electricity in accordance with the population's consumption of electricity. If the nuclear power plant's electricity network couples with hybrid energy sources such as the hydrogen power plant, wind, and so on, the load-following mode can adjust and control its generating electricity ratio commensurable to other stations' fraction. [23] Furthermore, depending on the weather condition, tropic countries' electric demand in the summer is more than in the winter; however, for high latitude area countries is reversed. Therefore, load-following can provide flexibility in generating electricity. For Europe, the European Utilities Requirements (EUR) establish the regulation that nuclear power plants must at least be capable of daily load cycling operation between 50% and 100% of their rated power with a rate of change of the electric output of 3-5% power per minute. Consequently, France follows this regulation and usually operates the reactors in a 12-3-6-3 daily loading cycle, meaning that reactor power maintains at 100% for 12 hours, decreasing power from 100% to 50% over 3 hours, following power by 6 hours at 50%, and rise back to 100% over 3 hours. [24, 26] Beyond that, the Giorgio Locatelli et.al research analyzes the economic aspect that SMRs, operating load-following couples hydrogen power plants with an Alkaline Water Electrolysis facility. Its research shows that a 100-50-100 daily operation cycle which operates reactor power at 50% staying for 12 hours (8 pm~8 am) that has more profitable compared to keeping power at 50% for 8 hours (0 am~8 am). [27] Moreover, Grünwald, Reinhard et.al [28] also consider applying about 100-50-100 daily operation strategy for PWRs in German, which can extend the nuclear power plants' lifetime and length of fuel cycles

When a reactor suffers slight malfunctions or trips, it may disconnect to the grid system and reduces the reactor power to the low level of around 20~30% of full power to supply the nuclear power plant auxiliary system instead of shutdown the reactor. This method keeps the reactor at power mode, so once the problem has been solved, the reactor can return to full power load quickly. In addition, it also provides an alternative electrical source that is different from the grid system and internal sources, including emergency diesel generators and batteries. [23] Moreover, when a reactor operates at a low power level, it is less detrimental on account of lower fuel temperature compared to loading full power. In that case, it means a reactor needs less coolant to maintain a safe status. Thus, owing to those reasons, it is important to understand the reactor transient behaviors at a low power level. This study will select 25% of power in the middle range of 20% to 30% power for the low power level behalf to comprehend the transient behaviors of i-PWRs in the low power level.

In summary, the base-load power change typically is 10% to 80%, and the flexible load-following power change usually is 20% to 80%. [23] Therefore, for the steady-state scenarios, this study selects operating reactor power at 100%, 75%, 50%, and 25% in 6, 12, 24, 48, and 72 hours. Additionally, the reactor power rate setpoint is 3% per minute. Even though running the reactor at 25% normalized power over a day is impractical on account of flux doubling, xenon poising, and so on, it is necessary to investigate and simulate the transient behaviors' trend. For the dynamic scenarios, this research selects and is based on France's 12-3-6-3 daily operation cycle and Giorgio Locatelli et.al research's 100-50-100 cycle that maintains power at 50% for 12 hours. All of the load following scenarios will comply with EUR regulations and under natural circulation operation to keep the inherent safety. For the sake of brevity, Table 4 presents the summary of scenarios.

Table 4 The overview of scenarios setup descriptions and parameters will be analyzed.

Category	Scenarios	Normalized power setpoint (%)	Operating time (hr)	Parameters	Annotation
Base-load					
	1	100	6	Generator load, Normalized power, Average generator load, Average normalized power, Range of generator load, Range of normalized power, Standard deviation of generator load and Standard deviation of normalized power	
	2	100	12		
	3	100	24		
	4	100	48		
	5	100	72		
	6	75	6		
	7	75	12		
	8	75	24		
	9	75	48		
	10	75	72		
	11	50	6		
	12	50	12		
	13	50	24		
	14	50	48		
	15	50	72		
	16	25	6		
	17	25	12		
	18	25	24		
	19	25	48		
	20	25	72		
Load-following					
	1	100-50-100	12-3-6-4	Generator load, Normalized power, Average generator load, Average normalized power, Range of generator load, Range of normalized power, Standard deviation of generator load and Standard deviation of normalized power	Based on France daily operation
	2	100-50-100	12-12		Based on Giorgio Locatelli et.al and Germany recommended daily operation

4. RESULT AND ANALYSIS

This section provides detailed explanations and results of reactor transient performance by executing the IAEA simulator. It presents the outcome for each scenario, including base-load and load-following cases with statistical data analysis and trends of the reactor parameters' alternations.

4.1 Base-Load Mode Scenarios

The result of normalized power under 100, 75, 50, and 25 % set points in 6, 12, 24, 48, and 72 operating hours are shown in Figures 15 to 34, presenting numerous oscillations of reactor transient behaviors. Numbersome factors cause normalized power oscillations. For example, the stochastic nature of the neutron chain reactions process includes delayed neutrons and random events that occur in the thermal-hydraulic transport process as heat transfers from fuel to the coolant. Furthermore, the statistical properties of the turbulent flow and flow induce control rods' vibration which makes the reactor's parameter values fluctuate. The energy and spatial dependence of neutrons, depicted in section 3 (Eq 3 and 6) also have an effect on fluctuations. With attention to the neutron leakage and scattering phenomena are illustrated in the neutron transport equation, contributing to reactor parametric values' oscillations. Moreover, neutron poison products are one of the factors that affect reactor performance. Generally, all of those aspects induce reactor power oscillation. On account of the variation of reactor power, those factors make moderator and coolant temperature fluctuations as reactivity feedbacks described by the temperature reactivity coefficients, including moderator and coolant. With those variations of moderator and coolant temperature, the steam generation rate will fluctuate. As a result, the deviation of the steam generation rate will lead to generator load fluctuations. To elucidate this research case, Table 5 summarizes the scenarios of base load cases.

Table 5 The overview of base-load scenarios

Scenarios	Brief description
1	Operate the simulator at 100% normalized power for 6 hours
2	Operate the simulator at 100% normalized power for 12 hours
3	Operate the simulator at 100% normalized power for 24 hours
4	Operate the simulator at 100% normalized power for 48 hours
5	Operate the simulator at 100% normalized power for 72 hours
6	Operate the simulator at 75% normalized power for 6 hours
7	Operate the simulator at 75% normalized power for 12 hours
8	Operate the simulator at 75% normalized power for 24 hours
9	Operate the simulator at 75% normalized power for 48 hours
10	Operate the simulator at 75% normalized power for 72 hours
11	Operate the simulator at 50% normalized power for 6 hours
12	Operate the simulator at 50% normalized power for 12 hours
13	Operate the simulator at 50% normalized power for 24 hours
14	Operate the simulator at 50% normalized power for 48 hours
15	Operate the simulator at 50% normalized power for 72 hours
16	Operate the simulator at 25% normalized power for 6 hours
17	Operate the simulator at 25% normalized power for 12 hours
18	Operate the simulator at 25% normalized power for 24 hours
19	Operate the simulator at 25% normalized power for 48 hours
20	Operate the simulator at 25% normalized power for 72 hours

4.1.1 Load Full Power Steady-State Scenarios

Figure 15 shows that the reactor operates for 6 hours to compare the fluctuations of France load following, which runs reactor at 6 hours in a temporal steady state and provides more details of oscillations' status. It demonstrates that the normalized power is about 99% initially and decreases to 97.6% after one hour. In the 2.5 hours, it reaches the minimum point and starts to slightly climb up to 97.7%. With attention to the Xe reactivity plot, the Xe reactivity reaches the maximum point in 2.5 hours, causing the normalized power declines. When normalized power decreases, it leads to the coolant average temperature becoming lower as presented in Figure 15. After 3 hours, the Xe reactivity finds the equilibrium, and the alternation of normalized power becomes relatively smaller compared to the beginning to the third hour. In addition, As the reactor feedbacks, the boron concentration in the reactor core decreases, which makes the normalized power and coolant average temperature increase. It can be observed those phenomena after 3 hours in Figure 15. Furthermore, the generator load corresponds to the normalized power trend.

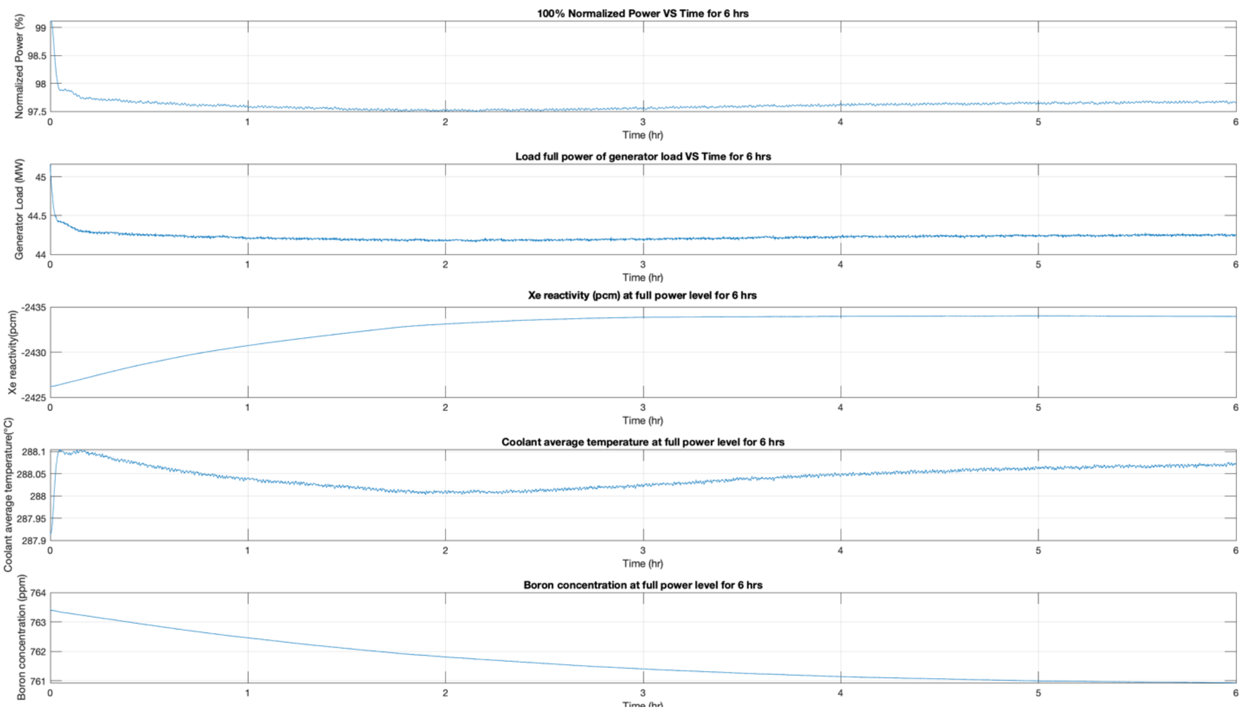

Figure 15. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature and boron concentration for Scenario 1 results.

Figure 16 shows operating 100% normalized power for 12 hours. For its first 6 hours, the trend is similar to Figure 15. Xe reactivity becomes more negative in three hours. As a result, the normalized power and generator rate decrease. Because when normalized power decreases, the production of heat decreases, causing the steam produces less. The curve of normalized power reaches the local maximum at the 7^{th} hour in Figure 16 and then starts going down on account of the accumulation of neutron poison. It shows a curve of the reactor normalized power at the time between the 3^{rd} to 12^{th} hour is symmetry and seems that the reactor tries to find the balance of its power via chemical shim as illustrated in Figure 16. Since the xenon reactivity and coolant average temperature decrease that leads boron concentration becomes less to compensate for those effects. The change of the coolant average temperature corresponds with normalized power.

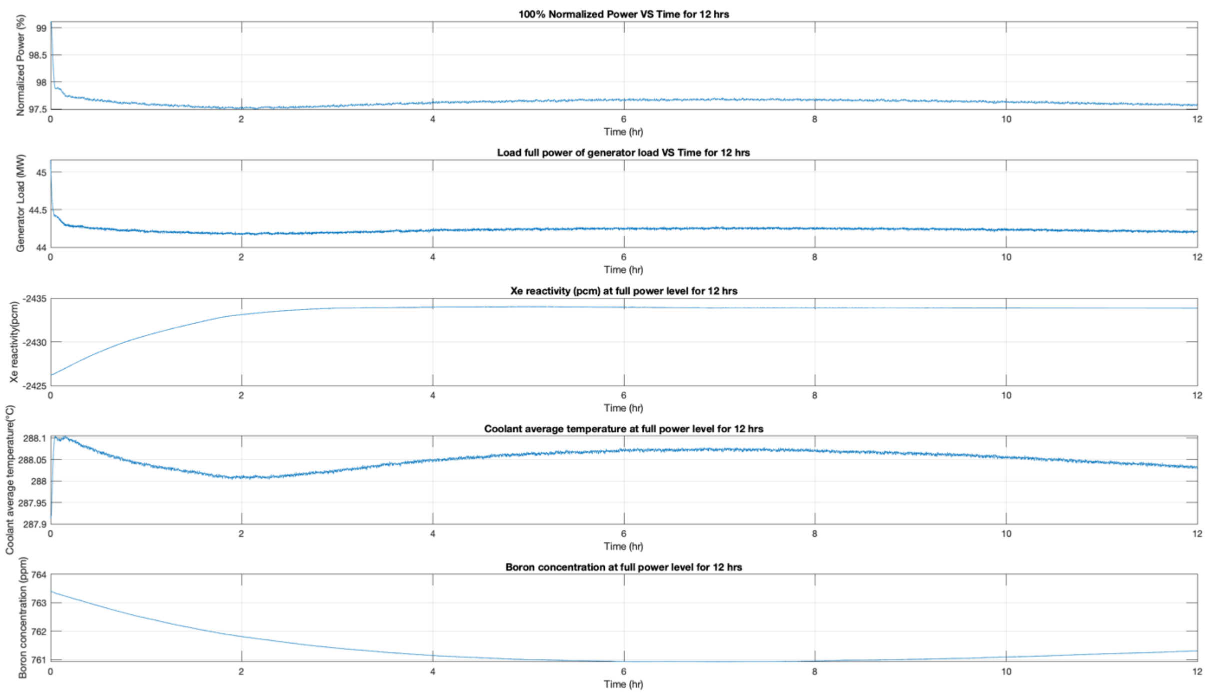

Figure 16. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature and boron concentration for Scenario 2 results.

Figure 17 is the result of operating the reactor for 24 hours at full power level. It indicates the reactor power's trend stops hovering in this range (97.5%~97.7%) after 12 hours and continues to decline in the interval of 10^{th} to 24^{th} hours due to the augmentation of boron concentration. Because when the coolant and fuel average temperature decrease as the reactivity negative feedback, it will add less negative reactivity to the moderator or fuel. Therefore, the boron concentration will increase to compensate for this change and keep the reactor in safety status. Additionally, the total amount of neutron poison products does not reduce and still keeps equilibrium between the decay rate of Xe-135 and the production of Xenon during complete burnup. As a result, the neutron production becomes less, and normalized power declines in Figure 17. Furthermore, the generator load also presents decreasing trend, which is relevant to the normalized power and coolant average temperature.

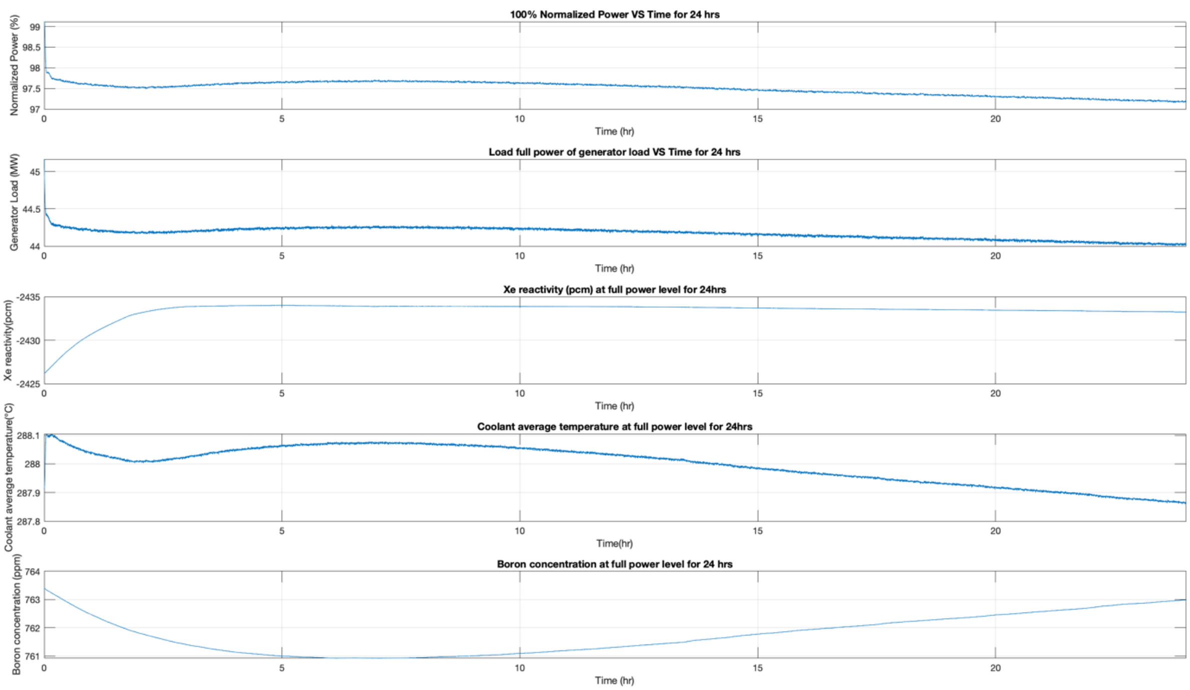

Figure 17. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature and boron concentration for Scenario 3 results.

Although this research focuses on the daily nominal operation situations, it is necessary to follow the transient behaviors and its trend after one day. Figure 18 shows the simulator runs for 2 days at the whole power level. For its first 24 hours, its outcome basically is similar to scenario 3, which has been discussed in the previous paragraph. After 24 hours, the Xe reactivity begins to increase due to the depletion of fuel. The coolant temperature still keeps dropping, and boron concentration continues going up. Considering the reactivity feedback, once the coolant and moderator temperature decrease, the fuel and moderator reactivity become less negative. Consequently, the reactor increases the boron concentration to maintain the reactor stability and comply with NRC 10 CFR Part 50 regulations, the reactor temperature feedback should be negative. As a result, the normalized power and generator load decline. Those decreasing patterns resemble scenario 3, where the reactor simulates for 24 hours at full load.

Figure 18 and Figure 19 present the trends and results for the simulator running at full power for 48 and 72 hours. For 48 hours case, the normalized power still drops from 97.5% to 96.5%, which decreasing trend is the same for 72 hours case, and it falls to 96% of normalized power. The generator load for scenario 4 also decreases to about 43.7%, and for scenario 5 drops to 43.4%, owing to the declining normalized power. The reasons for reducing normalized power are the depletion of fuel and the augmentation of boron concentration. Although the Xe concentration decreases slightly, it should make the normalized power augment. But on account of reactivity feedback and the coolant average temperature declines, leading to the reduction in reactivity of the fuel and moderator. To manage this alternation, the reactor uses chemical shim as boric water to maintain the stability of the reactor core. However, the reduction of xenon concentration makes the reactivity become less negative, which is the same result for the reactor feedback, so to keep the reactor inherently safe, the boron concentration still increases, as illustrated in Figure 18 and Figure 19. The trends accord to scenario 3 in Figure 17. In the end, the normalized power will keep decreasing and lead to the coolant average temperature declining as feedback increases the boron concentration, which induces the normalized power still fall until the fuel exhaust.

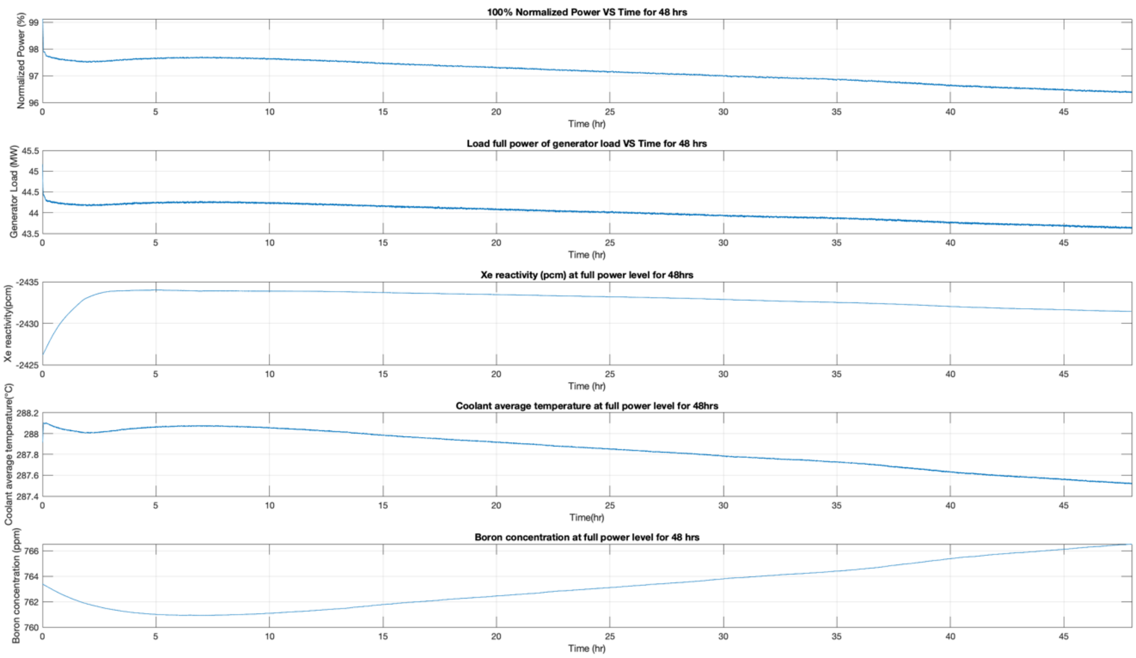

Figure 18. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature and boron concentration for Scenario 4 results.

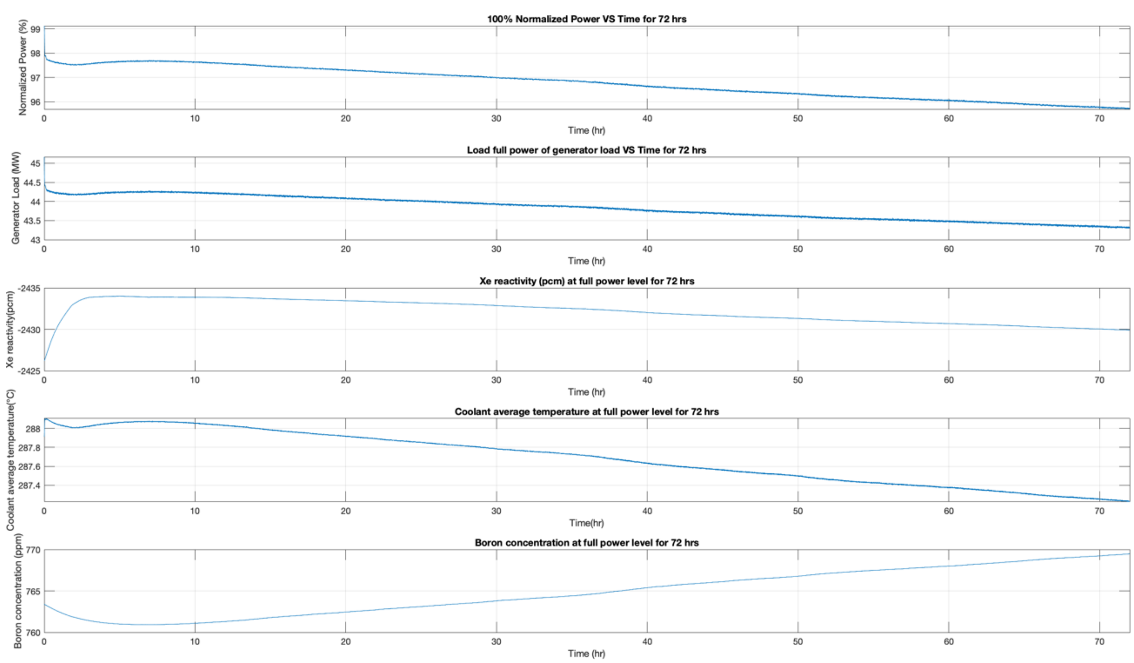

Figure 19. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature and boron concentration for Scenario 5 results.

4.1.2 Load 75% Power Steady-State Scenarios

Figure 20 presents the operating reactor at 75% normalized power for 6 hours. It can be observed that there are two large-amplitude oscillations at the beginning of an hour which is relative to the control rod's worth. In the first hour, xenon reactivity becomes more negative, meaning the xenon concentration increases. Meanwhile, in order to maintain the normalized power at 75% as this scenario setup, the control rods start to withdraw, and boron concentration slightly increases to compensate for the xenon influence, causing the fluctuation in normalized power, coolant average temperature, and generator load. This simulator rod's position control is based on the power setpoint and nuclear power when the simulator is in reactor leading mode. If the simulator's nuclear power is over the setpoint, the control rods start to insert and vice versa.

During the second hour, there are five smaller amplitude oscillations of normalized power and the time between each oscillation increases. The interval time of normalized power oscillations rises because the slope of the xenon increasing rate gradually decreases. Hence, the times of normalized power oscillations decline over time; however, each oscillation interval time arises gradually during the first to the sixth hour in Figure 20. Overall, the control rod is mainly to adjust the normalized power, and boron concentration plays a minor role in scenario 6. The generator load has the relevant trend of the normalized power. Because the normalized power is based on the reactor core status, once the normalized power increases, the fuel temperature increases as well. It will affect the production of steam. The kinetic energy of the steam is used in the turbine generator to produce electricity.

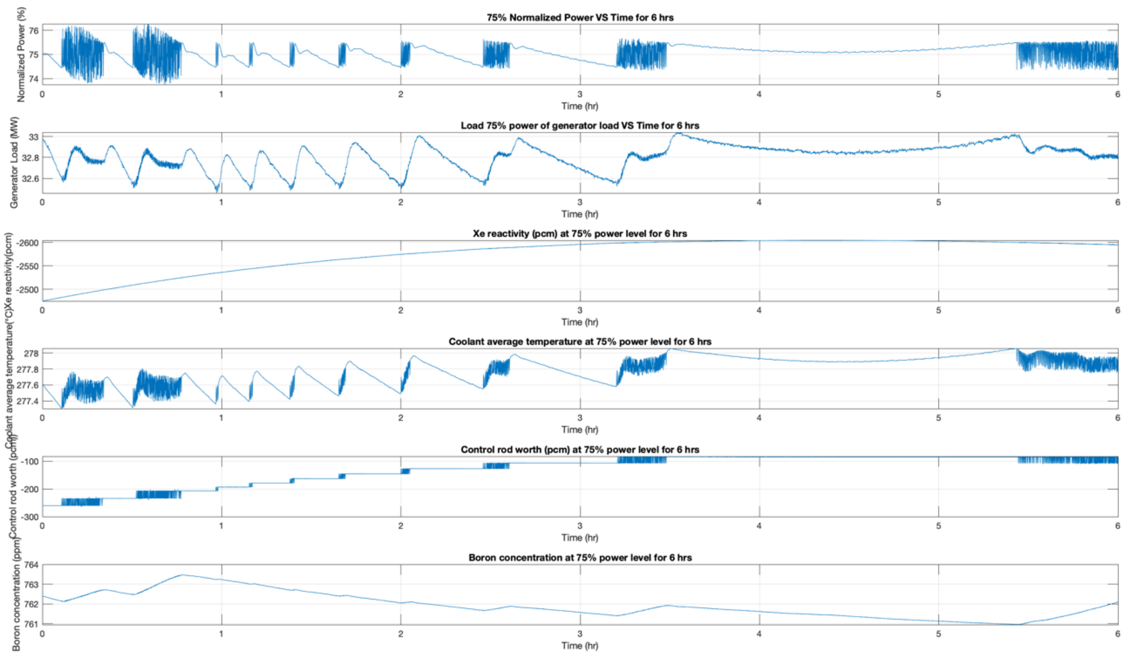

Figure 20. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 6 results.

Figure 21 demonstrates the result of scenario 7, and the pattern of its first 6 hours is similar to scenario 6. The interval time lengthens for each normalized fluctuation, which has been discussed in the previous paragrapher. However, this relationship stops at the 7th hour, as shown in figure 21. In addition, there is a long oscillation, showing from the fifth hour to the seventh hour. After that, the times of oscillations grow, and the interval of each oscillation is getting shorter, which shows the approximate symmetry trends in figure 21 between the first to the eleventh hour by reason of the xenon reactivity becoming less negative denoting xenon concentration decreases. Hence, with fewer neutron poison products, the normalized power will go up; however, to maintain 75% normalized power, the control rod starts to insert, and boron concentration increases. Meanwhile, the coolant average temperature decreases gradually because of reactivity feedback. Furthermore, it is important to notice that each oscillation corresponds to the control rod worth fluctuating in Figure 21. Moreover, the xenon-135 decay half-life is 9.2 hours, and the fuel does not burn up completely, so after about 7 hours, the xenon concentration drops

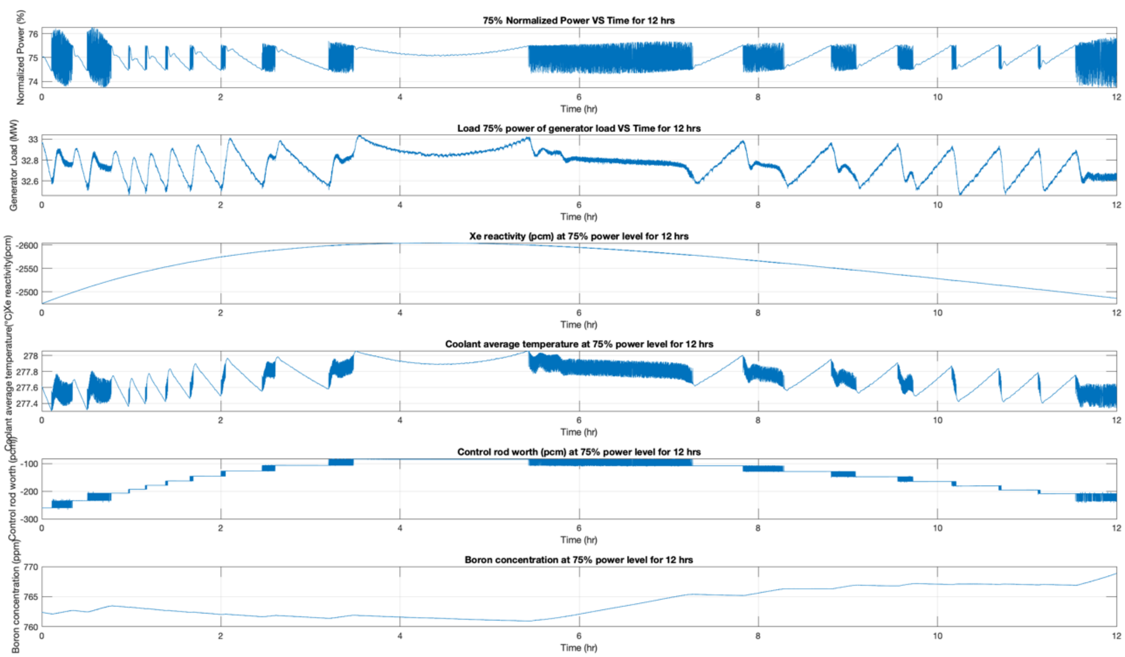

Figure 21. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 7 results.

Figure 22 presents the whole view of the normalized power oscillation from the 11th hour to about the 15th hour, which corresponds to the control rod worth fluctuating and presents an increasing trend due to decreasing xenon concentration. However, with negative feedback, the boron concentration increases, which starts the increasing trend after 15 hours. Besides, the fluctuation slightly decreases during the 11th hour to about the 15th hour. Nevertheless, it starts to oscillate from the 16th hour to the 24th hour in the range of 73.75% to 76% normalized power. On account of control rod worth oscillations, reduction of xenon concentration, and augmentation of the boron concentration. Those influences are reflected in the coolant average temperature and the generator load between 32.6 to 32.7 MW.

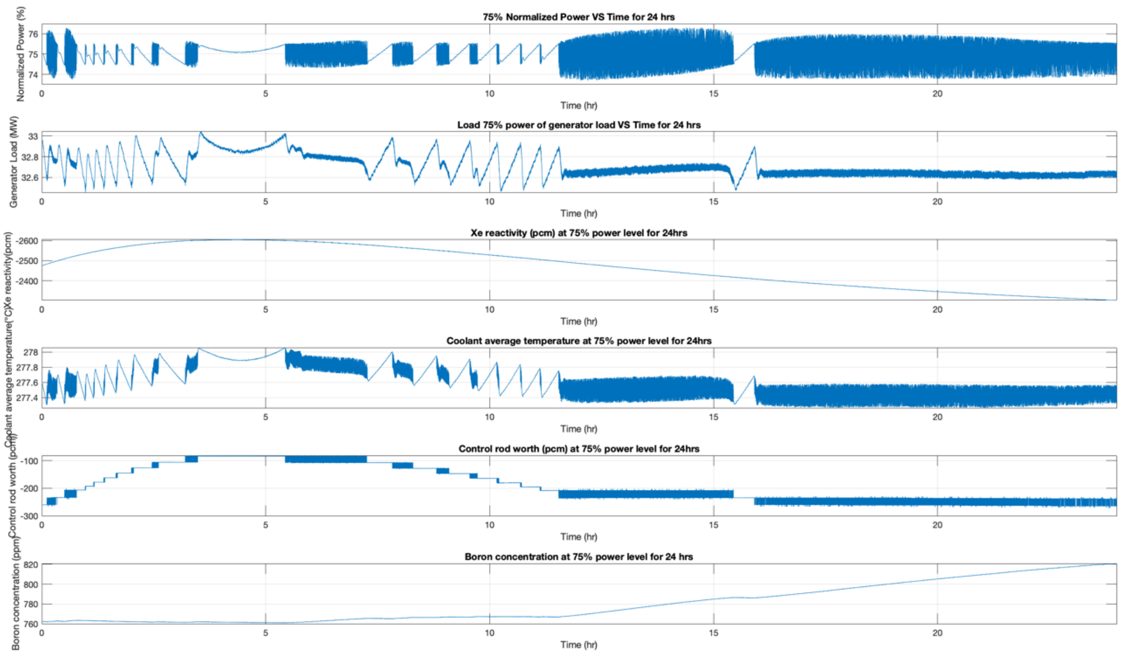

Figure 22. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 8 results.

Figure 23 demonstrates xenon reactivity reaches the equilibrium after 30 hours, causing the frequency of the control rod oscillations decreases, which reflects the normalized power, coolant average temperature, and boron concentration. As a result, the reactor core tends to be stable and does not show any obviously larger fluctuation after 42 hours. This trend continues for 72 hours, as illustrated in Figure 24. Although the amplitude of the 15th to the 25th hour is similar to the oscillations between the 28th to the 42nd hour, the generator load does not display resemble amplitude of the fluctuation because reactor transient behaviors need time to reflect it. The fluctuations between the 15th to the 25th hour are denser, compared to the oscillation of the 28th to the 42nd hour, denoting the reactor control more frequently between the 15th to the 25th hour leading to the generator load having a smaller range of fluctuations. The intervals of fluctuation from the 28th to the 42nd hour are longer, giving the reactor core more time to present its transient fluctuation. Figure 24 shows the end of the normalized, presenting the decreasing trend due to the loss of fuel, which is the same trend as scenario 5 in Figure 19. However, the simulator running at full power level presents this tendency earlier than 75% because the burnup ratio of 100% normalized power is higher than 75%.

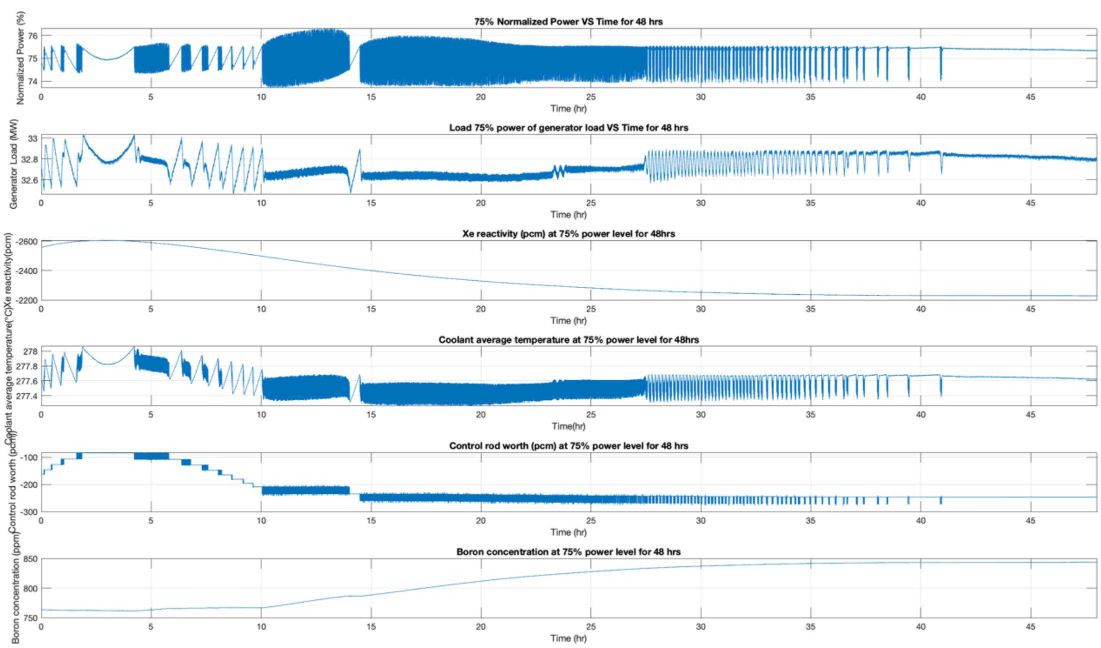

Figure 23. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 9 results.

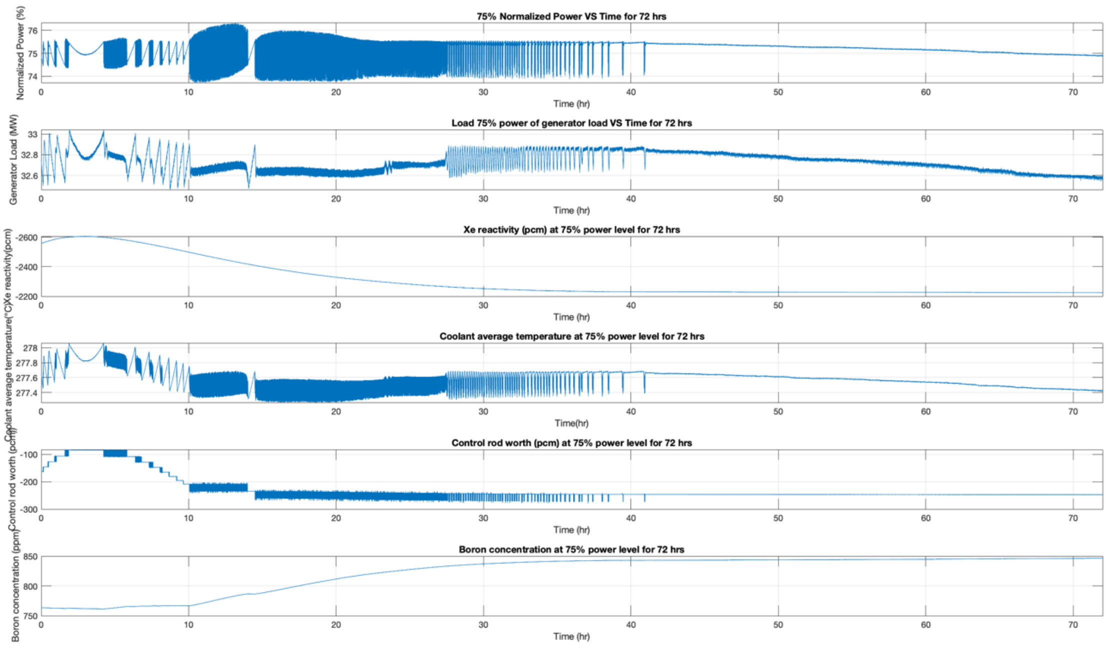

Figure 24. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 10 results.

4.1.3 Load 50% Power Steady-State Scenarios

When the reactor operates at 50% normalized power for 6 hours, there are three large amplitude oscillations at the beginning of thirty minutes in Figure 25. After that, the oscillations' amplitude becomes smaller, compared to those three oscillations. That trend is similar to 75% normalized power, which is illustrated in Figure 20. The xenon concentration increases, making the control rods withdraw, and boron concentration augments to maintain the normalized power at 50% of the setpoint. Because the boron concentration increasing slope rises that affects the control rod worth fluctuations, it makes the amplitude of normalized power becomes smaller at the first hour. Although the oscillations of 50 % and 75% normalized power have a similar trend initially, the times of 50% normalized power's oscillation is more than 75%, shown in Figure 25, due to the lower value of the normalized power setpoint. Additionally, the oscillation time also increases, and the interval of each oscillation grows, too. These relationships resemble 75% normalized power from the first to the third hour owing to the difference in Xenon concentration increasing slope ratio and the augmentation of the boron concentration as the chemical shim. The generator load is 20.7MW and decreases to an average of 20.2 MW in the first hour, which is dissimilar to the normalized power trend because the larger fluctuation of normalized power affects the steam production rate and coolant average temperature causing the coolant flow rate to variate. After that, the oscillations of the control rod worth correspond to the normalized power and the generator load.

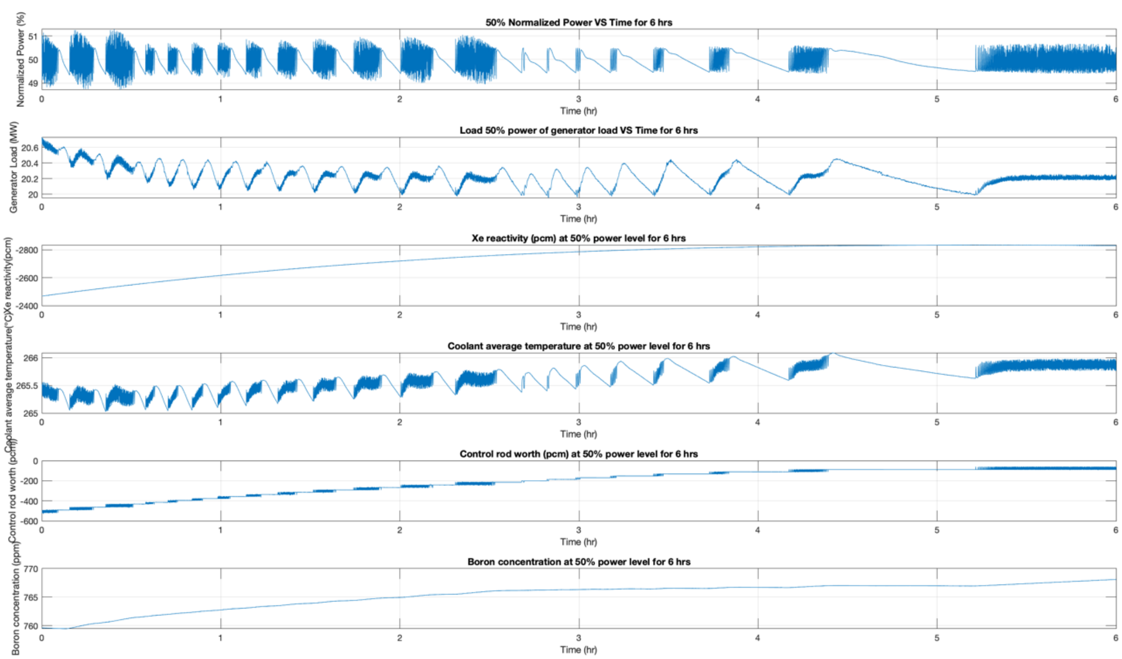

Figure 25. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 11 results.

Figure 26 has a similar pattern to figure 25 in 6 hours, and the curve is almost symmetry in the interval of the third to the ninth hour. Despite the 50% and 75% normalized power having the oscillation in about the sixth hour, the oscillation time for 75% normalized power is longer than 50% or normalized power owing to the xenon concentration and normalized power setpoint difference. At a low power level, the xenon concentration has a relatively large degree of influence. After 10 hours, the amplitude of oscillations suddenly becomes larger in Figure 26 due to the reduction of the xenon concentration ratio becoming larger. It makes the control rod worth fluctuating and the coolant average temperature decreasing as the reactor reactivity feedback makes the boron concentration augment. It can be observed that the pattern of the xenon reactivity trend and control rod worth are the same.

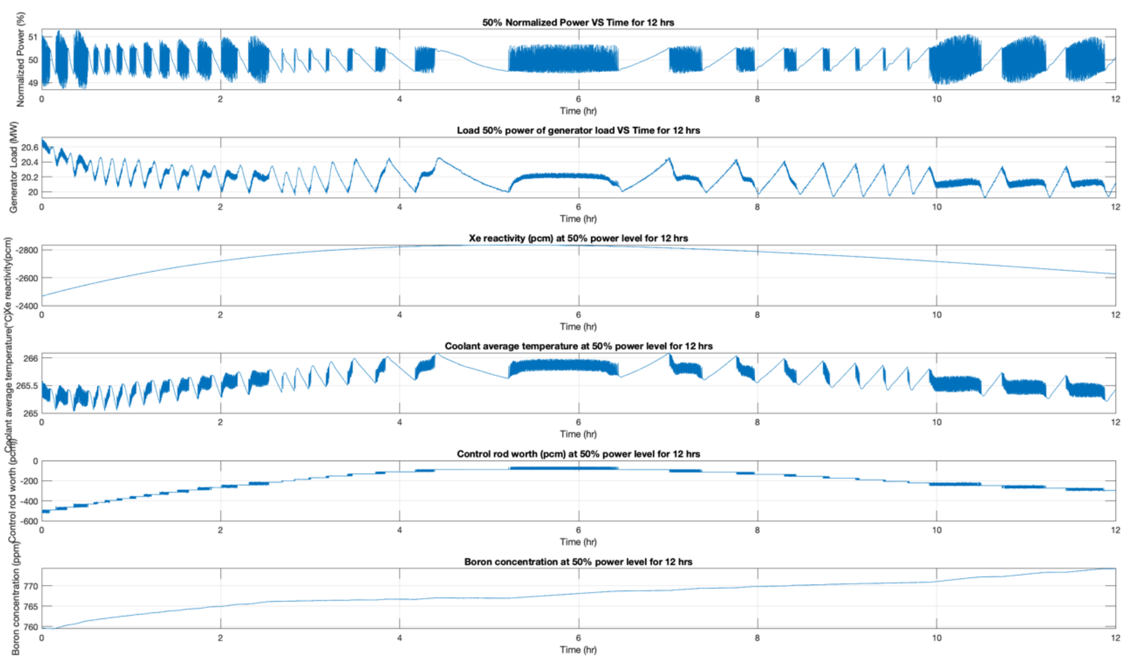

Figure 26. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 12 results.

Figure 27 presents that the normalized power oscillations turn into smaller amplitudes gradually from the 10^{th} hour to about the 15^{th} hour and the oscillation time decreases slightly. The reasons for that are the alternation of xenon reactivity and the change of the boron concentration. The boron concentration increases, which lessens the control rod worth oscillation ratio. Oscillations from the 15^{th} to the 24^{th} hour repeat the same pattern from the 10^{th} hour to about the 15^{th} hour in Figure 27. However, their amplitude of oscillations is larger than the time from the 10^{th} hour to about the 15^{th} hour because the reduction slope of xenon concentration increases, and the boron concentration goes up subtly. The reason for the boron concentration increasing slowly is that the coolant average temperature decreases gradually as the reactivity feedback, the fuel moderator reactivity becomes less negative, and the reactor releases boric water to maintain reactor safety. Once the reactor is close to the balance, the alternation rate becomes small.

Overall, the generator load in the first hour gradually decreases (Figure 25) and hovers at the range of about 20 MW to near 20.4 MW from the 2^{nd} hour to the 15^{th} hour in Figures 26 and 27. After the 15^{th} hour, the generator load starts to increase, as demonstrated in Figure 27. Additionally, due to large fluctuations of core reactivity between negative and positive, the coolant temperature at the core outlet decreases. As the feedback, it makes the coolant flow rate decline and the average fuel temperature will augment, leading steam flow to the turbine to increase. Consequently, the generator load shows an increasing trend. It is inversely to the 100% normalized power 24 hours scenario's trend and seems to need more time to achieve the balance point of the generator.

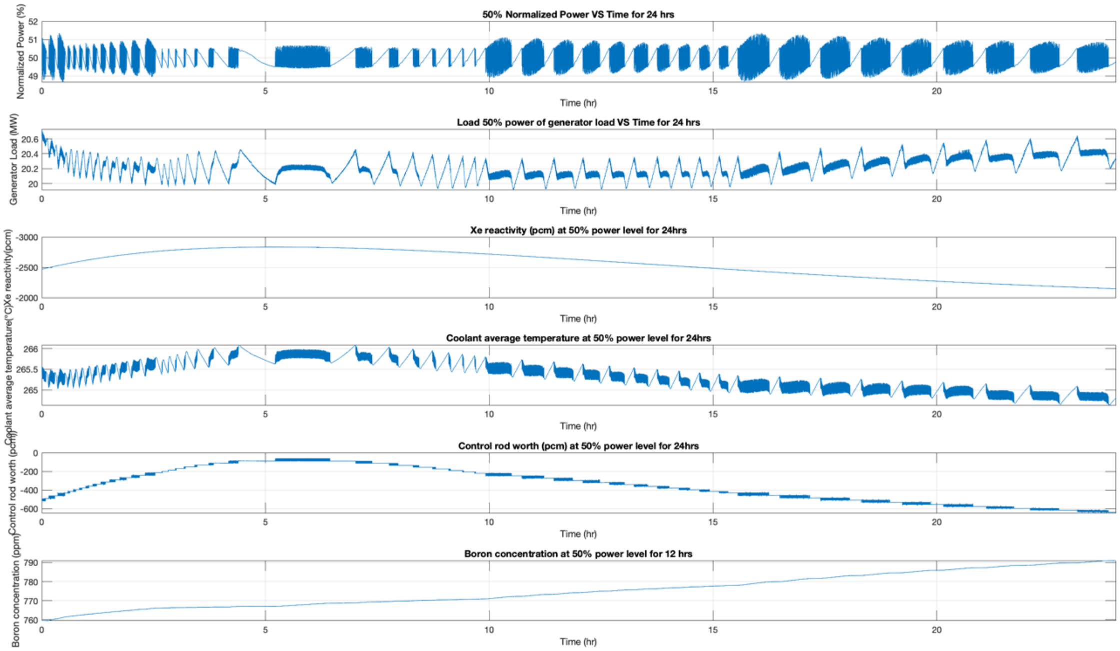

Figure 27. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 13 results.

Figure 28 shows the simulator runs at 50% normalized power for 48 hours. It can be noticed that normalized power after 25 hours has a larger amplitude of fluctuations because of the augmentation of boron concentration and oscillations of control rod worth. After 35 hours, those get to be stable. Therefore, the oscillations' amplitude becomes smaller, which corresponds to the generator load and coolant average temperature. This trend is similar to the 75% normalized power scenarios; however, the 50% scenario seems to postpone the outcomes from the 26th hour to the 45th hour in Figure 28 compared to the 15th hour to the 28th hour in Figure 23 due to the different setpoint of normalized power. The lower normalized power is more sensitive to xenon reactivity. Once the xenon reactivity is close to equilibrium, the fluctuation becomes less dense. However, the interval of each oscillation increases, giving the reactor's transient behavior enough time to demonstrate that temporarily increases the range of the generator load fluctuations in Figure 28 and Figure 29. It is important to notice that the eventual downtrend does not occur as in the scenarios of loading full and 75% normalized power. Due to lower normalized power, denoting the expense of fuel is not as much as theirs. Hence, although the decreasing trend does not show up in the result, it will present this trend as extending the simulating time.

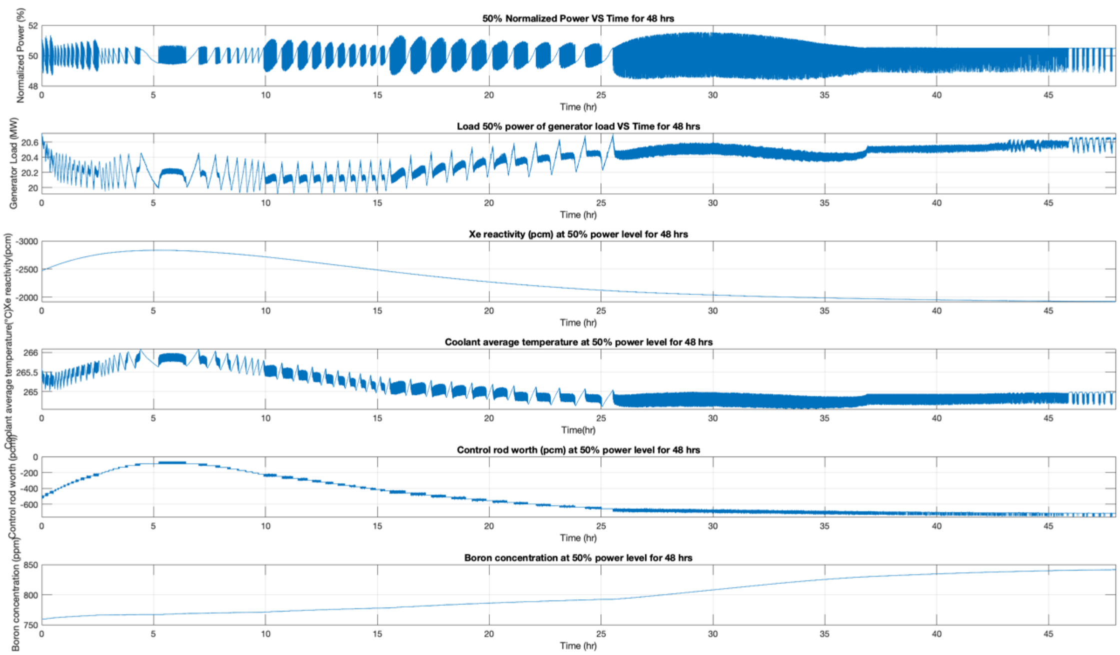

Figure 28. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 14 results.

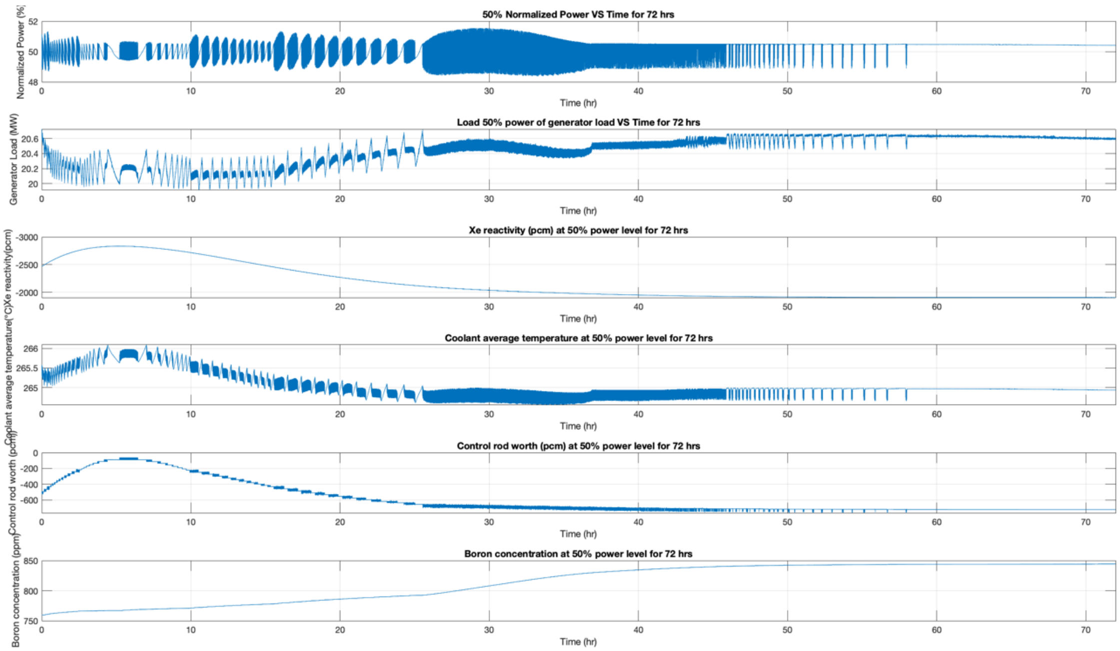

Figure 29. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 15 results.

4.1.4 Load 25% Power Steady-State Scenarios

Figure 30 shows the reactor operates for 6 hours at 25% normalized power. It is apparent to notice that normalized power markedly declines from approximately 25% to 12.5% after three hours which illustrates the IAEA simulator allows the operators have about three hours to fix the reactor trips or slight problems. After three hours, the reactor normalized power will drop on account of neutron poison and flux doubling, illustrated in Figure 30 xenon reactivity plot. Before the downturn, the oscillations kept occurring and hovering at 24% to 26% normalized power for three hours. The reactor starts to withdraw the control rods to overcome the neutron poison effect. Boron concentration slightly goes up owing to the reactivity feedback. However, even if the reactor completely withdraws all control rods, it still does not compensate for the xenon influence. As a result, the normalized power and coolant average temperature suddenly drops and do not arise back in 6 hours. The normalized power falls to 13%, and the generator load is from 9 MW to 3.5 MW.

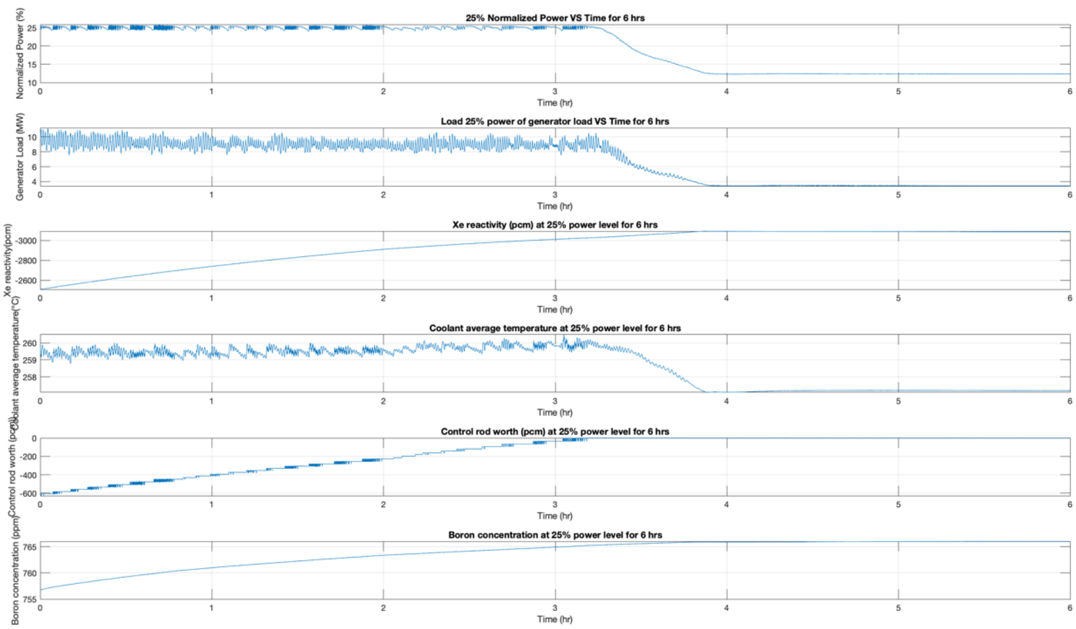

Figure 30. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 16 results.

Figure 31 presents the parametric value of normalized power, generator load, Xe reactivity, and so on for scenario 17. Before 6 hours, the result and trend are similar to scenario 16, that the neutron poison products such as Xe have a remarkable influence. Besides, the lower normalized power has more sensitive to xenon concentration. After 8 hours, the normalized power is still lower than the setpoint. But it should be noticed that the xenon reactivity becomes slightly less negative due to the xenon decay. Consequently, the normalized power slightly increases and corresponds to the generator load and coolant average temperature.

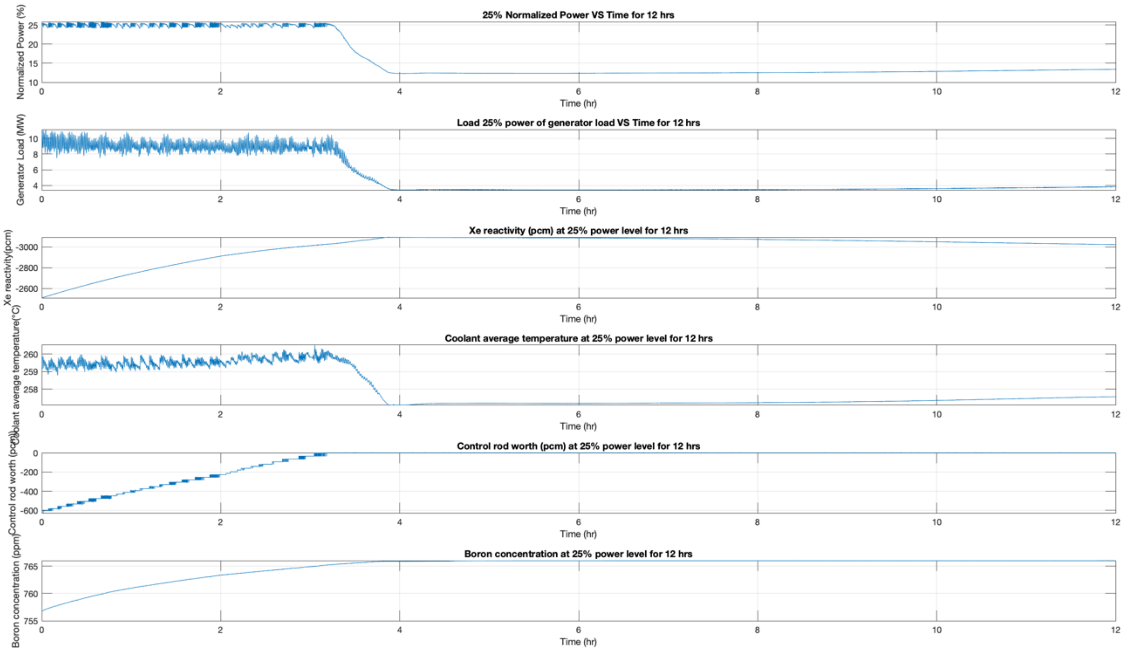

Figure 31. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 17 results.

Figure 32 demonstrates the simulator runs 24 hours at 25% normalized power. The normalized power is below setpoint status lasting for 12 hours until the xenon concentration drops. Once the xenon concentration decreases, the coolant average temperature and normalized power suddenly increase. Meanwhile, the control rods start to insert into the core to adjust the fuel reactivity to respond to the alternation of the neutron poisons. It can be observed that the xenon concentration becomes lower than in the beginning. As the reactor feedback, the boron concentration also begins to climb up at the 17th hour. The control rods insert more steps compared to the initial status, causing the generator load fluctuations to have larger amplitude, and the coolant average temperature presents decreasing trend. The pattern of the normalized power corresponds to the generator load.

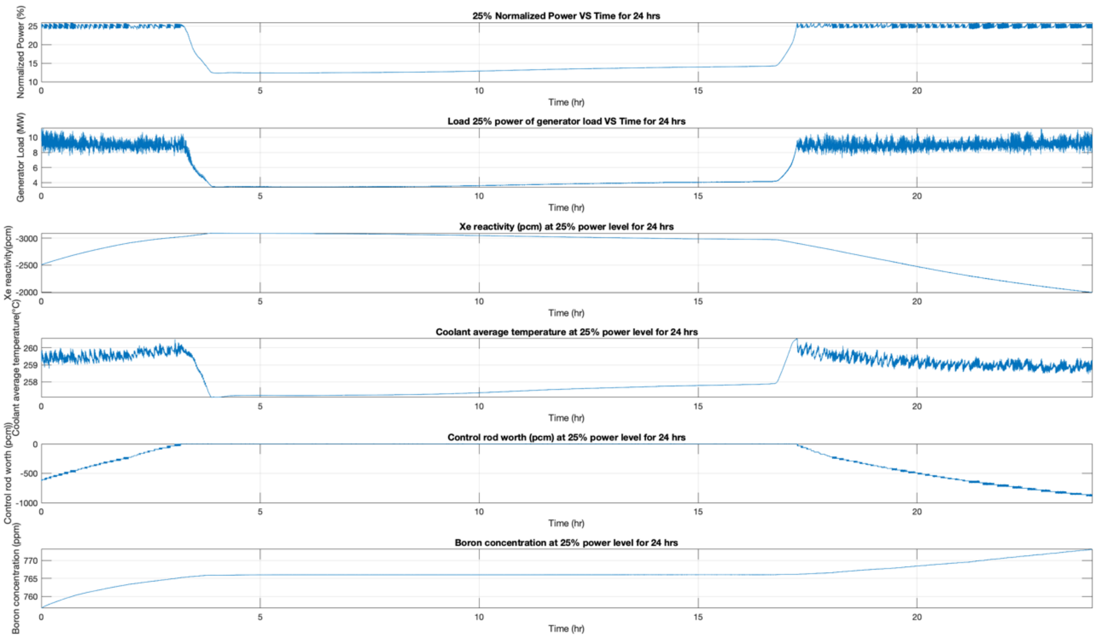

Figure 32. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 18 results.

The behavior and trend of the 25% normalized power steady-state scenarios are remarkably different from other steady-state scenarios because the normalized power at a low power level is more sensitive to xenon concentration. Figure 33 and Figure 34 show results that the simulator runs for 48 and 72 hours at 25% normalized power, separately. It can be observed that the xenon concentration continues to decrease after 24 hours due to the incomplete burnup at a low power level and the xenon-135 decay half-life. It makes the xenon concentration reduces more than the simulator operates at 50% and 75% normalized power. Eventually, the xenon reactivity reaches the equilibrium, demonstrated in Figure 33 and Figure 34, after 48 hours. Therefore, the control rod worth becomes comparably stable. The coolant average temperature, normalized power, and generator load have fewer fluctuations. In contrast, the boron concentration climbs up to help the reactor maintain balance conditions on account of xenon reactivity being less negative than the initial.

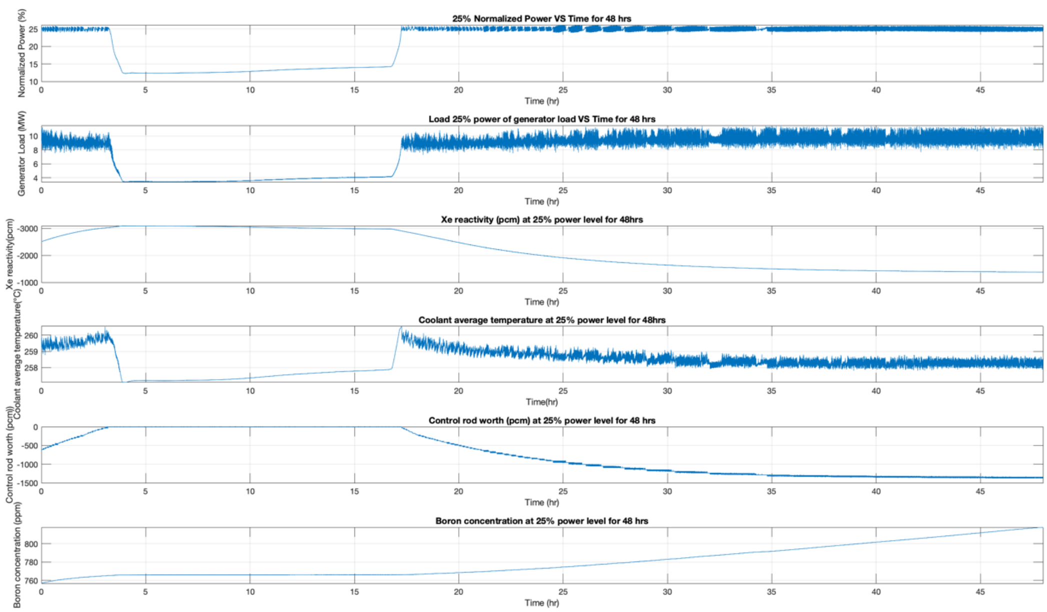

Figure 33. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 19 results.

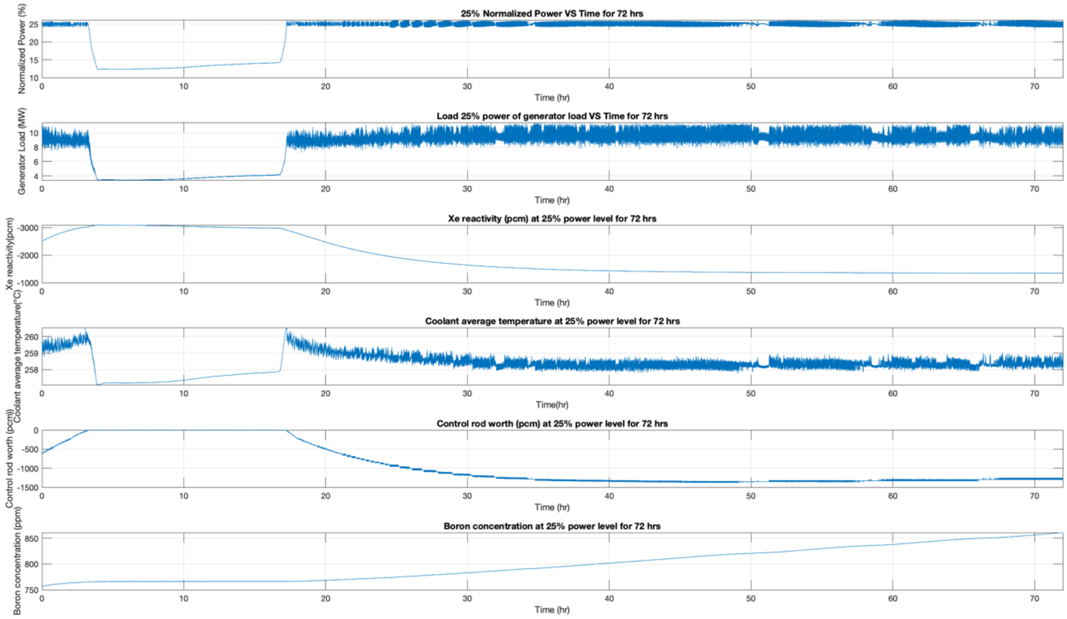

Figure 34. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Scenario 20 results.

4.1.5 The Statistical Result of Steady-State Scenarios

Tables 6 and 7 demonstrate the statistical result, including the range, average, and standard deviation for all steady-state scenarios. Table 6 displays that when the normalized power setpoints decrease, the ranges have an incrementing trend. This relationship correlates with the generator load side in Table 7, except for the generator load at 100% power level cases. Table 6 presents the amplitude of fluctuations for the 100% normalized power setpoint as the smallest compared to others. However, the generator load side does not correspond, as illustrated in Table 7. Except for the generator load at 25% power level cases, the generator load ranges at full power level are the biggest, and 75% power level is the smallest.

At the setpoint of 100% normalized power, the average power is about 97.6%, and the average power goes down to 96.771% in the case of operating for 72 hours. In contrast, the cases of 75% normalized power are from 75.0420% to 75.1544% when operating time increases to 72 hours on account of neutron poison effecting reactor transient behaviors. The 50% normalized power cases are the opposite trend of 100% normalized cases and are similar to 75% cases. When reactor operating time increases, the average normalized power slightly arises in 50% normalized power cases. The standard deviation for 100% normalized power cases has a larger value operating for 72 hours. However, 75% of normalized power cases have a bigger standard deviation for 24 hours situations. The 50% normalized power running 48 hours has a larger value of standard deviation. The normalized power at 25 % of cases has a maximum value at running 6 hours. Additionally, all generator loads are slightly less than the ideal generator load. On the normalized power side, only the average of 25% of normalized power cases does not reach the set point. The 25% normalized cases, owing to the drop after operating the reactor for 3 hours in Figure 30, show why 25% normalized power cases have the bigger range and standard deviation values. However, when increasing the operating time for 25% normalized cases, it can be found that the normalized power and generator load gradually go up. Its standard deviation becomes smaller. Compared to Tables 6 and 7, only normalized power at 100% and 25 % have strong relative to normalized power and generator load's range and standard deviation. For example, At the full and 25 percentage set points of normalized power cases have maximum values of standard deviation and range of generator load and normalized power for operating reactor 72 hours.

The maximum values of generator load's standard deviation usually are reactors operating 6 hours or 72 hours cases in Table 7. The range of generator load does not present any relationship.

However, the range of normalized power has a maximum value of either operating reactor for 48 hours or 72 hours. In brief, 100% normalized cases have smaller standard deviation values in the normalized and the generator load. For the range side, the normalized power at full power level, and the generator load at 75% power level have smaller range values. The normalized power of standard deviation and ranges values are larger than the generator load. Because when the reactor produces steam from the helical type of steam generator, the pressurizer acts as a cushion, and the main steam system valves will control the input of steam. Those keep the reactor pressure in the proper range, which diminishes the generator load fluctuations' degrees.

Table 6 The statistical normalized power results of steady-state scenarios

Setpoint of normalized power (%)	Time (hr)	Range (%)	Mean (%)	Standard deviation (%)
100	6	1.6232	97.6126	0.1064
	12	1.6213	97.6282	0.0825
	24	1.9621	97.5028	0.1744
	48	2.7536	97.1414	0.4095
	72	3.4162	96.7771	0.6523
75	6	2.5118	75.0420	0.3437
	12	2.5278	75.0248	0.3609
	24	2.5941	75.0044	0.5499
	48	2.6057	75.1703	0.4712
	72	2.5978	75.1544	0.4010
50	6	2.5849	49.9514	0.4103
	12	2.6334	49.9743	0.4144
	24	2.6859	49.9856	0.4576
	48	3.1650	50.0468	0.5910
	72	3.1703	50.1769	0.5275
25	6	13.5660	20.0192	5.9461
	12	13.5709	16.8194	5.7291
	24	13.6537	18.2907	5.7824
	48	13.8350	21.6530	5.3168
	72	13.8264	22.8423	4.5773

a. The values are based on the IAEA simulator digital results.

Table 7 The statistical generator load results of steady-state scenarios

Setpoint					
Normalized Power (%)	Ideal Generator load (MW)	Time (hr)	Range (MW)	Mean generator load (MW)	Standard deviation (MW)
100	45	6	1.0084	44.2241	0.0507
		12	1.0125	44.2315	0.0388
		24	1.1647	44.1745	0.0806
		48	1.5545	44.0028	0.1946
		72	1.8704	43.8276	0.2997
75	33.75	6	0.5800	32.8051	0.1224
		12	0.5819	32.7693	0.1212
		24	0.5866	32.7100	0.1109
		48	0.5737	32.7475	0.0984
		72	0.5795	32.7287	0.0936
50	22.5	6	0.7778	20.2318	0.1331
		12	0.8121	20.1901	0.1283
		24	0.8158	20.2211	0.1340
		48	0.8024	20.3588	0.1817
		72	0.8061	20.4472	0.1909
25	11.25	6	7.8123	6.7790	2.6482
		12	7.7331	5.3714	2.5531
		24	7.8462	5.9908	2.5788
		48	8.1013	7.7749	2.6048
		72	8.0450	8.3930	2.2937

a. The values are based on the IAEA simulator digital results.

4.2 Load-Following Mode Scenarios

Figure 35 is the France load-following operation result. Table 8 is the statistical result of load-following scenarios and analyzes the local steady-state oscillations. In order to follow France's daily operation as much as possible and comply with EUR regulation, this case operates the reactor for 25 hours which is slightly longer than one day. Because when withdrawing all control rods and increasing reactivity causes a prompt jump that can be noticed in Figure 35, the normalized power is 50% and rises to 60%. Owing to this reason, the power change rate can not increase more than the built-in time delay of precursors. As a result, after rising to 60% normalized power, the slope of the normalized power is apparently different. Meanwhile, all control rods have already been withdrawn. After the 12th hour, inserting control rods makes the reactor power goes down to 50%, causing vibration phenomena induced by flow and reactor feedback effect of alternation of fuel, moderator, and coolant temperature. In the meantime, the total flow rate and PZR level decrease.

The normalized power is about 97.6 % when the reactor power setpoint is at full power level as demonstrated in Figure 35. Besides, the fluctuation range and standard deviation of 100% normalized power as the setpoint are smaller than the setpoint of 50% normalized power, as shown in Table 7. This trend also corresponds to the generator load result. In addition, the amplitude of normalized power oscillations is subtly larger than the generator load in Figure 35. When the reactor's normalized power decreases to 50%, the reactor has more fluctuations at the beginning on account of the reactor feedback effect as variations of moderator and coolant temperature. Furthermore, when normalized power declines to 50%, xenon concentration augments due to incomplete burnup. Therefore, the control rods changes with neutron poison, and boron concentration slightly increases to cooperate with control rods for fear that fuel reactivity exceeds the regulation limit becoming too positive.

Figure 36 presents Giorgio Locatelli et.al and Germany's recommended daily operation result as the 100-50-100 cycle. After the 12th hour, the normalized power oscillations' standard deviation is smaller than at the beginning of 12 hours in Figure 36. Furthermore, the amplitude of normalized power is larger than the generator load. This result is corresponding to the France daily operation result. In order to follow this scenario and adhere to EUR regulation, the power rate sets at 5% as the maximum value for EUR requirements. Figure 36 shows the prompt drop situation as inserting control rods immediately at the 12^{th} hour, and after that, the power change rate becomes slow determined by the neutron precursor groups. The alteration reasons for xenon reactivity and boron concentration are as same as France's load-following scenarios.

The average 50% normalized power and generator load of France's daily operation for six hours is slightly less than Germany's recommended operation for 12 hours. For 100% of normalized power, both values of the normalized power and the generator load are similar. For the range side, the generator load at 50% setpoint power level of Germany recommending operation has a remarkably large value, compared to France operation scenario in Table 8. Beyond that, Table 8 shows the range values of 50 % are larger than 100% no matter in France or Germany recommended setups that are correlative to the normalized power of steady-state scenarios as illustrated in Table 6. For standard deviation, it can be noticed that 100% normalized power setpoint result of the normalized power and the generator load are noticeable smaller than 50 %. This relationship also matches the steady-state result in Tables 6 and 7. Moreover, the normalized power of range and standard deviation have larger values, which are also similar to the steady state results in Tables 6 and 7, owing to the function of the pressurizer and main steam valves to adjust the steam input.

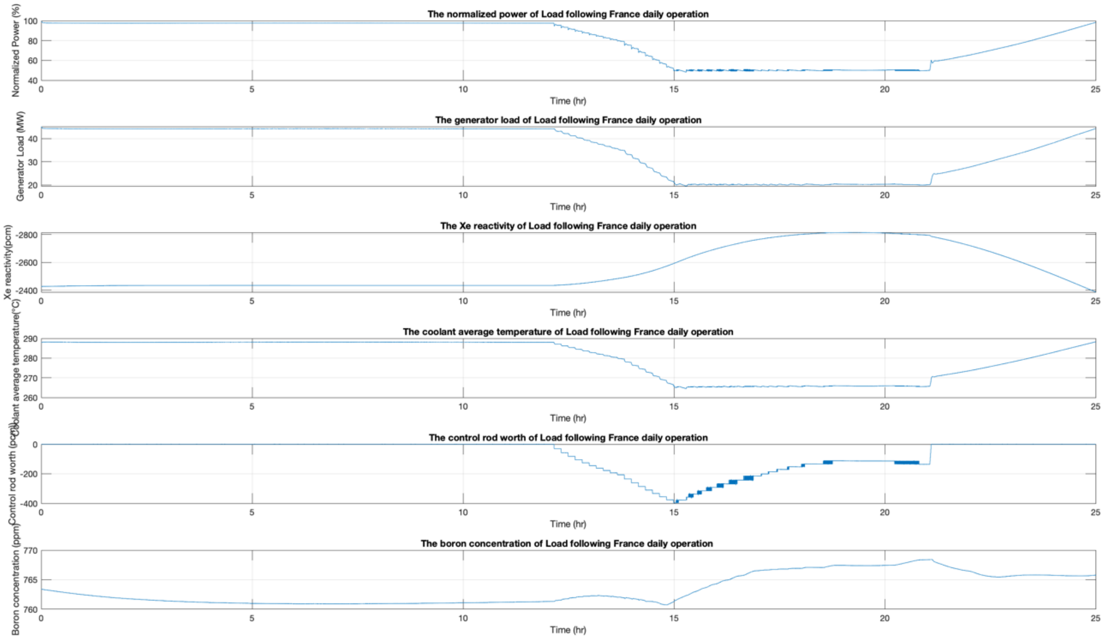

Figure 35. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for France load-following operation

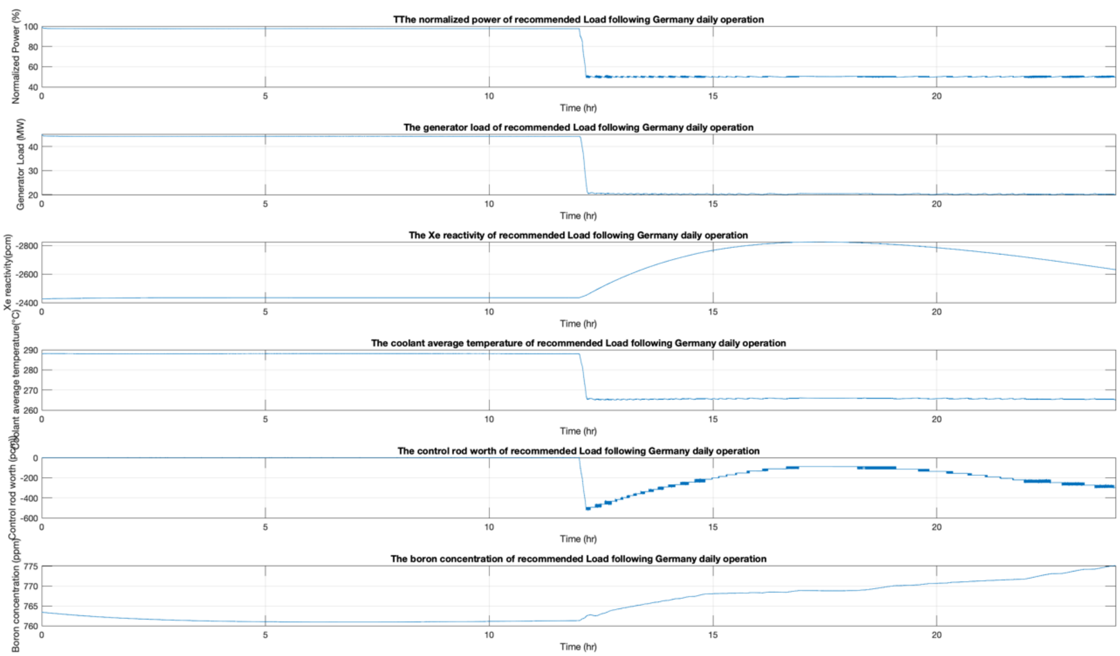

Figure 36. Plots from top to bottom are normalized power, generator load, Xe reactivity, coolant average temperature, control rod worth, and boron concentration for Germany recommended load-following operation

Table 8 The statistical normalized power and generator load results of load-following scenarios

Type of daily operation	Operating time	Normalized power setpoint	Average		Range		Standard deviation	
France			Normalized power (%)	Generator load (MW)	Normalized power (%)	Generator load (MW)	Normalized power (%)	Generator load (MW)
	6	50%	49.9924	20.2199	2.957	1.7748	0.3927	0.1365
	12	100%	97.6280	44.2314	1.662	1.0123	0.0818	0.0389
Germany								
Recommend	12	50%	50.0254	20.2259	2.7181	5.4993	0.4141	0.2130
	12	100%	97.6249	44.2301	1.6227	1.0070	0.0848	0.0407

a. The values are based on the IAEA simulator digital results.

4.3 The Overview of the discussion

In the base-load cases, when the reactor operates at full normalized power level, the large influence on reactor transient oscillation is xenon concentration, unlike other base load cases where the control rods mainly cause the fluctuations. Because when the reactor is at 100% normalized power, the control rods fully withdraw. Hence, the 100 % normalized power cases' standard deviation, no matter in normalized power or generator load in Tables 6 and 7, is relatively smaller than other base-load cases.

For 75% and 50% of normalized base-load cases' trends are similar, and their oscillations correspond to the alternation control rod worth. On account of different power level settings, it can be observed that 50% of cases have delayed response. Generally, both cases have correspondingly standard deviations of generator load and normalized power. Additionally, it is important to realize when increasing the operation time, the standard deviation of normalized power and generator load may increase or decrease depending on which power level operates. For instance, the generator load's standard deviation at 75 % normalized power level decreases when extends the operating time. In contrast, 50 % normalized power cases' standard deviation of generator load increases when operating time increases. Therefore, with this concept in mind, it is necessary to investigate the best settings and sensitivity during the operation for each reactor type.

The 25% normalized cases' performance is different than other cases due to operating the reactor at a relatively lower power level and xenon concentration suppressing reactor power presentation. Thus, the standard deviation of normalized power and generator load at 25% normalized power are larger than others. The load-following cases' statistical results of normalized power and generator load conform to base-load results and trends. The 100% power level operation settings have smaller standard deviations, and when extended operation time at 50% normalized power, the standard deviation increases, which corresponds to 50% normalized power base-load cases. Hence, whether operating in base-load or load-following modes, that would not cause a huge difference to the reactor's normalized power and generator load deviations which is important to know when operating the small modular reactor with a small or micro electricity grid.

5. CONCLUSIONS

The goal of this research aims to analyze performance of integral small modular reactor configurations through evaluations of operational characteristics if an integral PWR system. This analysis has been performed using the IAEA small modular reactor simulator. A number of transient scenarios have been evaluated considering nominal daily operation situation such as base-load and load-following modes. This research focused on the reactor's several parameters such as normalized power, generator load, control rod worth, coolant average temperature, and xenon reactivity. Fluctuations of normalized power and generator load have been assessed as a result of changes in those operational parameters. The methodology uses the IAEA SMR simulator as an analysis device of the i-PWR transient performance simulation tool to investigate and assess the oscillations. This section summarized main conclusions resulting from this analysis.

5.1 Summary

Owing to the small modular reactors usually connecting to small or mini electricity grids, the small fluctuations may cause huge influences on the grid system, making it unstable. In order to maintain the stability and reliability of the grid system, it is necessary to evaluate the small modular reactors' dynamic behaviors and presentations. Therefore, to assess the IAEA i-PWR simulator's transient performance, this research selects normalized power at 100%, 75%, 50%, and 25% to operate the reactor under steady-state for 6, 12, 24, 48, and 72 hours separately, and dynamic operating conditions. All input operation values comply with the EUR regulations. The total scenarios are

22 to research the sensitivity of normalized power and generator load under different systems conditions.

This research finds out the alternation and oscillations of normalized power are relative to control rod worth and xenon reactivity parametric data on account of neutron poison accumulations and reactor negative feedbacks, which can be verified in the coolant average temperature data in section 4. In addition, when the simulator operates at the lower normalized power, it presents more sensitivity to neutron poison products such as xenon-135 at a lower power level. Hence, the normalized power at 100%, 75%, and 50% cases trends are more similar compared to 25% normalized cases. The 50% cases show the reactor delayed responding trends as well. Its decreasing normalized power tendency is not as remarkable as in 100% and 75% cases. The 25% of normalized power cases are deeply affected by neutron poisons, so their normalized power suddenly drops after operating for 3 hours. However, after about 12 hours, the 25% cases' normalized power rises owing to the decay of xenon.

The fluctuations of generator load are corresponding to the normalized power because the steam production is relevant to the reactor power level and heat. However, the range of the generator fluctuations is smaller than the normalized power on account of the main steam system valves adjusting and the pressurizer, which acts as a steam cushion. Thus, the variation of normalized power can not cause larger amplitudes compared to the normalized power data. The standard deviation of normalized power becomes bigger when increasing simulation time, except for 25% normalized power cases, which is similar to the generator load trend. When extends the operation time, the range of normalized power augments except for 25% and 75% cases. Although 25% and 75% normalized power do not follow this trend, their normalized power's range maximum values are operating for 48 hours and are only slightly bigger than 72 hours values.

This study's load-following scenarios have two types. One is France's daily operation, and the other is Germany and Giorgio Locatelli et.al recommendation operating mode. Both operation modes' results resemble steady-state scenarios. The range values of normalized power and generator at lower power levels are larger than at higher power levels. Besides, the standard deviation at 50% normalized power demonstrates becoming larger as increasing operation time, which has the same tendency as the steady-state scenarios. The France daily operating at 100% normalized power for 12 hours has a smaller standard deviation of normalized power and the generator load compared to the Germany recommendation. However, both normalized power and the generator range and average values are similar.

5.2 Future Research

This research accomplishes the goal to demonstrate the IAEA i-PWR simulator transient statistical result and tendency. It discusses the reasons causing fluctuations and analyzes sensitivity with relevant parametric values that presents the view of the i-PWR dynamic behaviors. However, this study uses the default value of boron concentration which is not realistic for the operation situations. Hence, future work will try to simulate under zero boron concentration as original water to appropriate to the pertinent actual circumstances. Besides, utilizing research data results analyze the reactor transient fluctuations affecting the mini electricity grid. To improve the accuracy, it is necessary to evaluate the reactor noise and errors of this simulator.

Other areas for future research will couple with different types of power plants, such as hydrogen power plants to simulate the small modular reactors cooperating in the real-world benefits and economy. Another aspect is to modify the design applied to the desalination plant, the production of radiopharmaceutical or boron neutron capture therapy.

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