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Nuclear Hybrid Energy System Southeast Regional Case Progress Report

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October 2018

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ORNL/TM-2018/1052

Reactor & Nuclear Systems Division

NUCLEAR HYBRID ENERGY SYSTEM SOUTH EAST REGIONAL CASE PROGRESS REPORT

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ACRONYMS

ARIS	Advanced Reactors Information System
BOP	balance of plant
CHP	combined heat and power
CPU	central processing unit
DOE	US Department of Energy
HP	high pressure
HPC	High Performance Cluster
IAEA	International Atomic Energy Agency
IP	industrial process, intermediate pressure
LP	low pressure
NHES	nuclear hybrid energy system
ORNL	Oak Ridge National Laboratory
PHS	primary heat system
PI	proportional integral
RAVEN	Risk Analysis Virtual Environment
TCV	turbine control valve
TRANSFORM	TRANsient Simulation Framework Of Reconfigurable Models

ABSTRACT

This report discusses ongoing work to develop and advance the modeling framework for evaluating the nuclear hybrid energy system (NHES) concept from a dynamic performance and cost perspective. A high-level multitiered steam header / turbine step-down system used in many chemical plants was coupled with a multimodular nuclear reactor facility representative of potential NHES deployment scenarios in the southeastern United States to begin demonstrating the market feasibility of these hybrid systems. The preliminary results of this system in three different scenarios is presented herein, and future work is discussed.

1. INTRODUCTION

The southeastern region of the United States has historically been friendly to nuclear power projects, due in large part to the regulated electricity market, coupled with the business models of the Tennessee Valley Authority and Southern Company. In addition to a favorable market structure, the southeast has large amounts of available cooling water, growing populations/economies, and a large variety of thermal energy–intensive industrial operations such as chemical and paper/pulp facilities. These and other aspects of the southeastern region provide motivation to investigate the potential market of nuclear hybrid energy systems (NHESs) in this region.

This report discusses preliminary results of the application of a multimodular nuclear facility that provides process steam and electricity cogeneration with a focus on the models developed. Because this work is preliminary in nature, it is conceptually high-level. As part of ongoing work, it will include more details and economic analysis.

1.1 ASSOCIATED TOOLS

The NHES program uses a variety of tools. The principal components of the modeling work are described briefly below.

1.1.1 NHES Repository

The NHES repository is located on the internal Gitlab server of the Idaho National Laboratory's High Performance Cluster (HPC). The repository contains the Modelica models and RAVEN workflows/modules specific to the NHES project. All work is committed according to proper procedures contained in the NHES repository wiki page to ensure proper version control.

1.1.2 Modelica

Modelica [1,2] is a nonproprietary, object-oriented, equation-based programming language used to conveniently model complex physical and cyberphysical systems (e.g., systems containing mechanical, electrical, electronic, hydraulic, thermal, or control, components). A key advantage of Modelica is its separation of physical models and their solvers. This separation enables rapid generation of complex physical systems and control design in a single language without requiring deep knowledge of numeric solvers, code generation, etc.

1.1.3 TRANSFORM

The TRANSient Simulation Framework Of Reconfigurable Modules (TRANSFORM) [3] is a component library developed at Oak Ridge National Laboratory (ORNL) using the Modelica programming language to investigate dynamic thermal-hydraulic systems and other multiphysics systems. The TRANSFORM library has been successfully used for a variety of nuclear applications, including investigations into the performance of NHESs [4,5], liquid metal [6] and gas-cooled reactors [7], and molten salt applications, including kinetic behavior and fission product transport [8–10].

1.1.4 Reliability Module

ORNL has developed a time-dependent reliability framework to minimize reliability-related costs over lifecycle costs (operational and maintenance [O&M] costs). The intent is for this capability to enable for the designer/analyst to assure reliable operation of an NHES system from early in the design phase. The reliability model of the NHES, which is written in Python, combines the stochastic processes of degradation and fluctuating load which are combined to evaluate time-dependent reliability. The component degradation process is modeled with piecewise Weibull distribution by using operational parameters (valve position, flow rate, etc.) from the Modelica model to estimate the Weibull parameters in every selected time step. Details of the model and implementation of the approach for a selected component—a turbine control valve (TCV)—are given in previous work [11].

1.1.5 RAVEN

The Risk Analysis Virtual Environment (RAVEN) [12] is a platform designed to perform parametric and probabilistic analysis based on the response of complex system codes. RAVEN can investigate the system response and the input space using a variety of methods (e.g., Monte Carlo) and has a complex statistical analysis framework for several types of applications (e.g., uncertainty quantification and sensitivity analysis). For more details on the use of RAVEN for the larger NHES work, see Rabiti et al. [13].

2. SOUTHEAST REGIONAL CASE DESCRIPTION

The southeast case selected is based on the Eastman Chemical plant in Kingsport, Tennessee. At their Kingsport facility, Eastman operates 17 boilers fueled by coal and natural gas. These boilers feed into a complex network of 19 steam turbines which serve the dual purpose of generating electricity for the facility and providing steam at the appropriate pressure and temperature to run chemical processes throughout the facility (Figure 1). This type of system is a combined heat and power (CHP) plant. The Eastman CHP generates an average of 155 MWe, all of which is consumed onsite. The steam generation rate is 3,600,000 lb/hr (454 kg/s) at a variety of pressures. The steam is used to provide process heating, to power air compressors and refrigeration machines, and to provide space heating inside the facility.

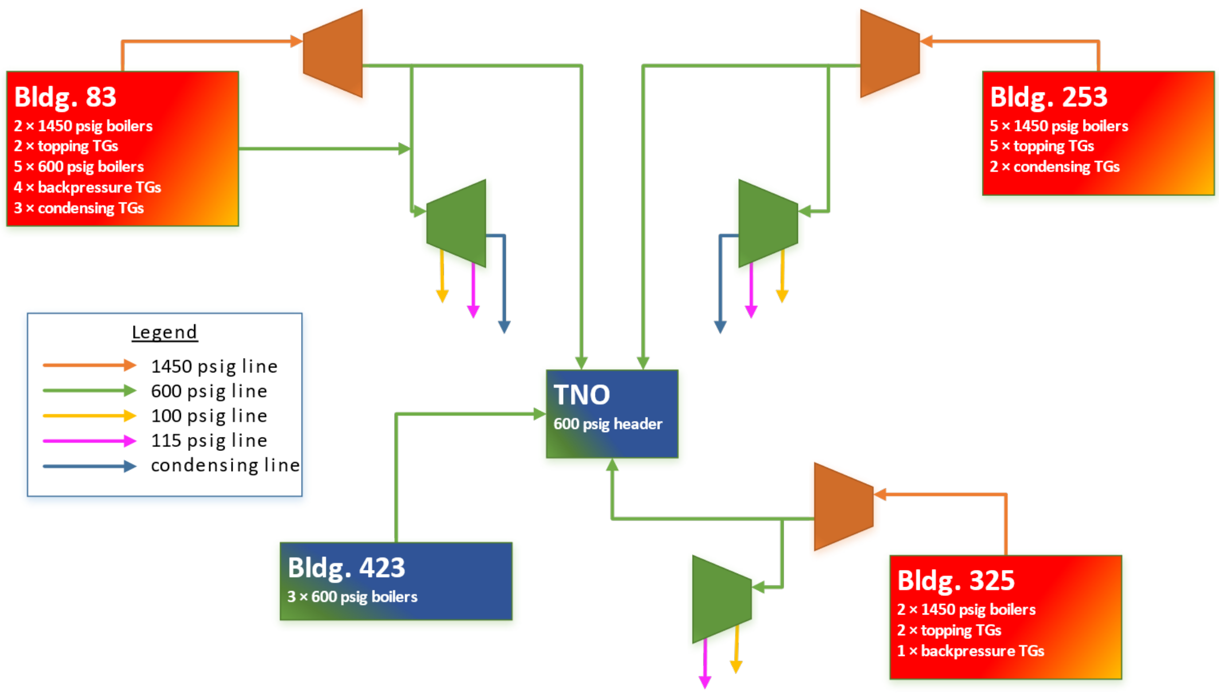

Figure 1. Process diagram of the existing Eastman combined heat and power plant [14].

As the intent of this study is to investigate the application of nuclear energy in a system such as Eastman's, and due to the lack of details at the CHP plant, this preliminary work created a nuclear system in which multiple smaller units work in tandem to produce steam coupled with a very simple steam distribution system. Future work will focus on creating a steam distribution and turbine system that is more representative of a complex CHP plant like Eastman's.

2.1 MODELICA MODELS

The following subsections describe the nuclear reactor and steam header / turbine system Modelica models generated for this report as they currently stand. Additional subsystems such as the energy manifold, balance of plant (BOP), and battery are consistent as presented in related reports [15]. These models are contained in the NHES repository discussed in Section 1.1.1.

2.1.1 Generic Multimodule Nuclear Reactor

The nuclear reactor model that was generated for multimodule applications is roughly based on open-source information. The general dimensions, layout, and systems of the plant were based on the NuScale plant as described in an International Atomic Energy Agency (IAEA) Advanced Reactors Information System (ARIS) database and open literature [16,17]. The fuel geometry was taken from a Westinghouse report [18], as that information was not available in the NuScale documents. Additional estimations such as details of the steam generator and forced flow in place of natural convection flow were made due to lack of details and to improve numeric robustness. A general description of the model's parameters are shown in Table 1 and Table 2. The Modelica implementation of the reactor is shown in Figure 2.

Table 1. Overall dimensions and operating conditions of the generic modular reactor model

Component	Length [m]	Diameter [m]	Parameter	Value
Reactor vessel	20	2.75	Power [MWt]	160
Inlet plenum	2	2.75	Pressure [MPa]	12.76
Core	2	1.5	Hot leg temperature [°C]	325
Outlet plenum	3	2	Cold leg temperature [°C]	285
Hot leg	10.5	1.4	Core temperature rise [°C]	40
Pressurizer	2.5	2.75	Mass flow rate [kg/s]	700
Steam generator	5.5	2.75/1.4	Steam pressure [MPa]	3.5
Cold leg	12	0.75	Steam inlet temperature [°C]	300
			Steam outlet temperature [°C]	200
			Steam mass flow rate [kg/s]	75

Table 2. Steam generator and reactor core parameters of the generic modular reactor model

Steam generator parameter	Value	Reactor core parameter	Value
Heat exchanger type	Helical coil	Radius of fuel rod [m]	0.004572
Outer tube diameter [m]	0.0127	Cladding thickness [m]	0.000571
Tube thickness [m]	0.0021082	Pellet to cladding gap [m]	7.87E-05
Tube pitch to diameter ratio	1.5	Fuel pellet radius [m]	0.003922
Number of tube passes	2	Fuel rod pitch [m]	0.012598
Number of tubes	2,240	Assembly size	17 × 17
Tube length	26.0752182	Fuel rods per assembly	264
		Non-fuel rods per assembly	25
		Number of assemblies	38

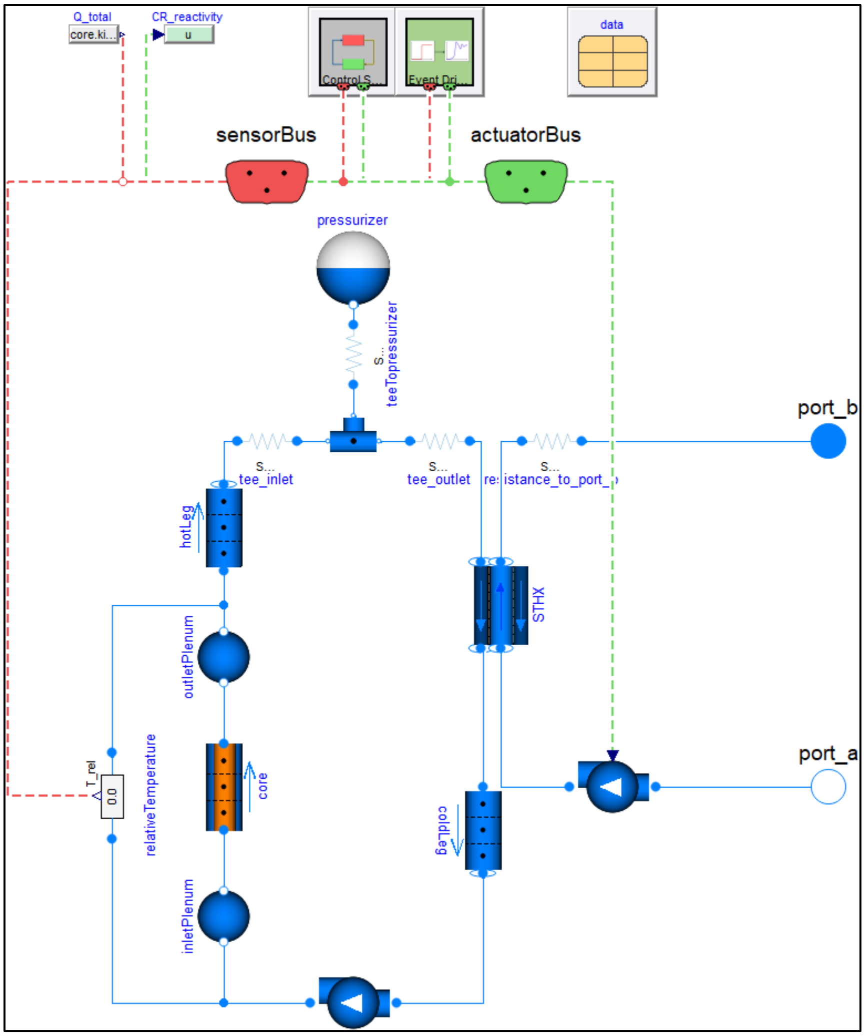

Figure 2. Modelica implementation of a generic modular nuclear reactor.

2.1.1.1 Reactor Physics

The reactor core model includes traditional point kinetics models, including decay heat groups, fission product behavior, and reactivity feedback. The parameters for each of the groups are shown in Table 3 through Table 5. Precursor and decay heat group data are taken from the TRACE manual [19] and fission product data from Nuclear Reactor Physics textbook and the Chart of the Nuclides [20,21].

Table 3. Neutron precursor group parameters. $\beta_{eff} = 0.006488$

Neutron precursor group	Decay constant [1/s]	Fission yield [α]
1	3.87E+00	2.60E-02
2	1.40E+00	1.28E-01
3	3.11E-01	4.07E-01
4	1.15E-01	1.88E-01
5	3.17E-02	2.13E-01
6	1.27E-02	3.81E-02

Table 4. Fission product parameters

Fission product	Decay constant [1/s]	Absorption cross section [b]
Tritium	log(2)/1.69e8	0
^{135}Te	log(2)/19	0
^{135}I	log(2)/23760	0
^{135}Xe	log(2)/32760	2.60E+06

Table 5. Decay heat group parameters

Decay heat group	Decay constant [1/s]	Generation fraction of fission power
1	1.77E+00	2.99E-03
2	5.77E-01	8.25E-03
3	6.74E-02	1.55E-02
4	6.21E-03	1.93E-02
5	4.74E-04	1.16E-02
6	4.81E-05	6.45E-03
7	5.34E-06	2.31E-03
8	5.73E-07	1.64E-03
9	1.04E-07	8.50E-04
10	2.96E-08	4.30E-04
11	7.58E-10	5.70E-04

2.1.1.2 Control System

Three control system options are currently implemented for use with the modular reactor. The first is the no controls option. In this approach, flow rates are held constant, and passive reactivity feedback mechanisms drive the reactor power. The second option is the steady state control option. This control system attempts to hold reactor thermal power and average temperature increase across the core constant based on nominal, constant setpoints. The third method holds the average core temperature increase constant via a constant setpoint, but it allows the reactor's thermal power to fluctuate based on a time variant setpoint to the proportional integral (PI) controller. This variable setpoint is established based on the user's discretion for the application. For the integrated system simulations presented in the report, the change in the power setpoint ($\Delta Q_{setpoint}$) from nominal power is based on changes to the nominal flow rates ($\dot{m}$) of steam in the energy manifold (i.e., industrial processes and energy storage) and BOP. This is illustrated in Eqs. 1–3 below. The minimum and maximum operations are to limit the change of the setpoint to avoid overheating/cooling issues that may lead to numerical or other physical issues (i.e., pressures and temperatures being too low or too high). The two active control systems are shown in Figure 3.

$$\Delta Q_{setpoint} = \min\left(1.05, max\left(\frac{Q_{BOP,setpoint}}{Q_{BOP,nominal}}\alpha_{BOP,nominal} + \frac{\dot{m}_{Other}}{\dot{m}_{Other,nominal}}\alpha_{Other,nominal}, 0.5\right)\right) \quad (1)$$

$$\alpha_{BOP,nominal} = \frac{\dot{m}_{toBOP,nominal}}{\dot{m}_{fromAllPHS,nominal}} \quad (2)$$

$$\alpha_{Other,nominal} = \frac{\dot{m}_{toOther,nominal}}{\dot{m}_{fromAllPHS,nominal}} \quad (3)$$

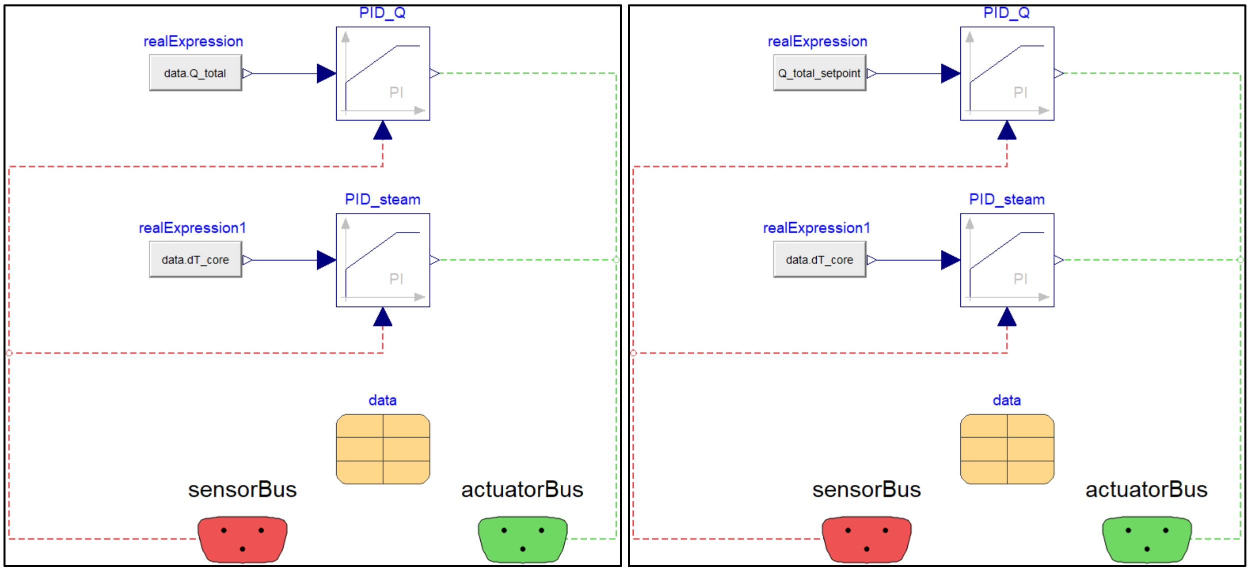

Figure 3. Simple steady state (left) and load following (right) control options; the difference lies in the setpoint definition for the core reactivity controller (PID_Q).

2.1.2 Steam Header and Step-Down Turbine System

For the simplified steam header and step-down turbine system, an industrial process (IP) was generated to demonstrate the ability to draw steam at various locations and to begin exploring feedback on the nuclear reactor systems to which it is connected. The implemented model consists of two steam turbines in series with mixing volumes between from which time-dependent flow rate demands remove steam, mimicking loads on the steam distribution network. In the current implementation, an idealized pump sets the pressure at the high pressure (HP) steam header and allows the flow rate to vary based on the system dynamics. Also, an ideal heat source is added to the HP header to keep the nominal temperature constant. Nominal parameters for the system are shown in Table 6, and the Modelica implementation is shown in Figure 4. Future work is will significantly expand this type of system to better mirror a complex CHP such as that of Eastman (Figure 1). LP indicates low pressure.

Table 6. Nominal flow rates and pressures of the simplified steam header and step-down turbine system

Parameter	Mass flow rate [kg/s]	Pressure [MPa]
To IP	200	4.13
HP steam process	20	4.13
To HP turbine	180	4.13/0.68
IP steam process	18	0.68
To IP turbine	162	0.68/0.1
LP steam process	16	0.1
To BOP	146	0.1

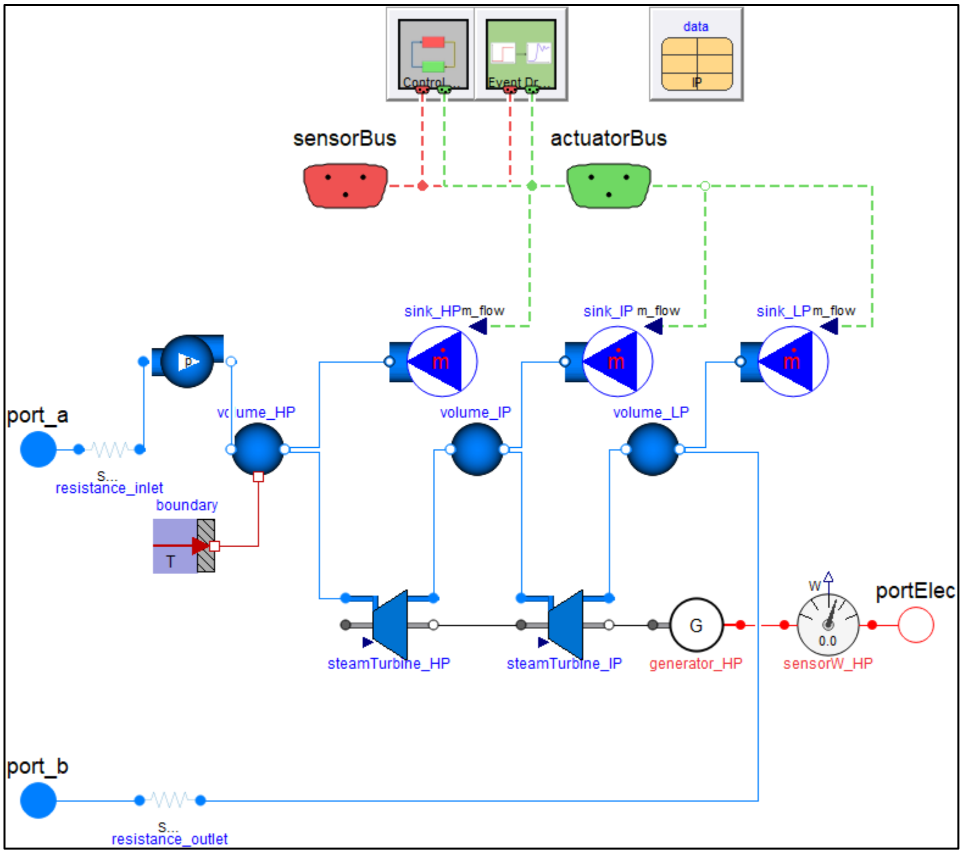

Figure 4. Simplified steam header and turbine step-down system.

2.2 RELIABILITY MODELS

This subsection summarizes the NHES reliability modeling work underway for the southeast regional case emphasizing the required modifications for the component and subsystem level time-dependent reliability model. The reliability module is as described in Section 1.1.4. The regional case design configuration differentiates from previous work configuration because the PHS includes multi-input SMRs (three-units modeled) instead of one SMR and load-following operation mode will change the stochastic load and therefore, TCV reliability is expected to decrease. Section 2.2.1. briefly discusses a dynamic reliability model requirement for multi-unit systems and investigates the dependencies between units. Section 2.2.2. focuses on the updates of the component reliability model.

2.2.1 Multi-Module Nuclear Reactor Reliability

The events at the Fukushima nuclear power station highlights to the need for consideration of risks from multiple nuclear reactor units co-located at a site. As a result, considerable research efforts have been dedicated to addressing the multi-unit risks over the past few years. Most of the technical problems faced in evaluating multi-unit risk are caused by dependencies between units. These dependencies such as common initiating events / shared systems, structures and components, shared instrumentation, control, fiber optics, other cables, electric divisions hared systems (e.g., FPS) and capacity of shared equipment (e.g., batteries) should be captured by developing a dependency matrix [22]. The PHS in the system design is therefore further decomposed to create the dependency matrix and subsystem interactions will be captured using non-Markovian Stochastic Petri Nets previously developed for the BOP.

2.2.2 Component Reliability under Load-Following

Failure modes of the TCV were identified, and the reliability model was defined according to the failure mode of the component. The reference failure rate was assumed as 2.5E-2/demand, which represents the failure of the TCV to open/close. This rate is reported in the component reliability database [23]. Only functional failure is being considered in this work, which is a failure to support a process need (flow of fluid, provide electrical power, etc.). Failure mechanisms such as stress corrosion cracking are not considered since current simulation capabilities have not yet included failure mechanisms. This approach complies with the US Nuclear Regulatory Commission's maintenance rule [24], which includes a performance measure based on functional failure.

The shape parameter (β) and scale parameter (η) in the Weibull model are used in reliability equations to determine lifecycle qualities of the data sets over time (t). The corresponding probability density function is given by

$$f(t|\beta,\eta) = \frac{\beta}{\eta^{\beta}} t^{(\beta-1)}\, exp\left\{-\left(\frac{t}{\eta}\right)^{\beta}\right\}, \text{for} \;\; \eta > 0, \text{and } \beta > 0$$

It is expected that in case of load following operational mode, β parameter will be increased, the estimated Weibull β values imply wear-out conditions (at the bathtub curve) and the characteristic life η and mean time between failures will be decreased based on simulated valve positions.

3. PRELIMINARY RESULTS

This section discusses the scenarios simulated and a corresponding discussion of the results. As this work is preliminary, specific conclusions are not drawn.

3.1 SCENARIO DESCRIPTION

Three scenarios were run to test the robustness of the implemented models and to form a preliminary understanding of the behavior of a multimodular nuclear reactor facility coupled with time-variant demand constraints. The three scenarios are summarized in Table 7. Each of these scenarios retains the same physical model layout and parameters and only modifies the control system of the nuclear reactors or primary heat systems (PHSs). The definition used for a control state is discussed in Section 2.1.1.2. The overall system being simulated is shown in Figure 5.

Table 7. Simulation scenario descriptions

Scenario	Description
1	No control for PHSs A, B, and C
2	Steady state control for PHSs A, B, and C
3	Steady state for PHSs A and B and load following control for PHS C

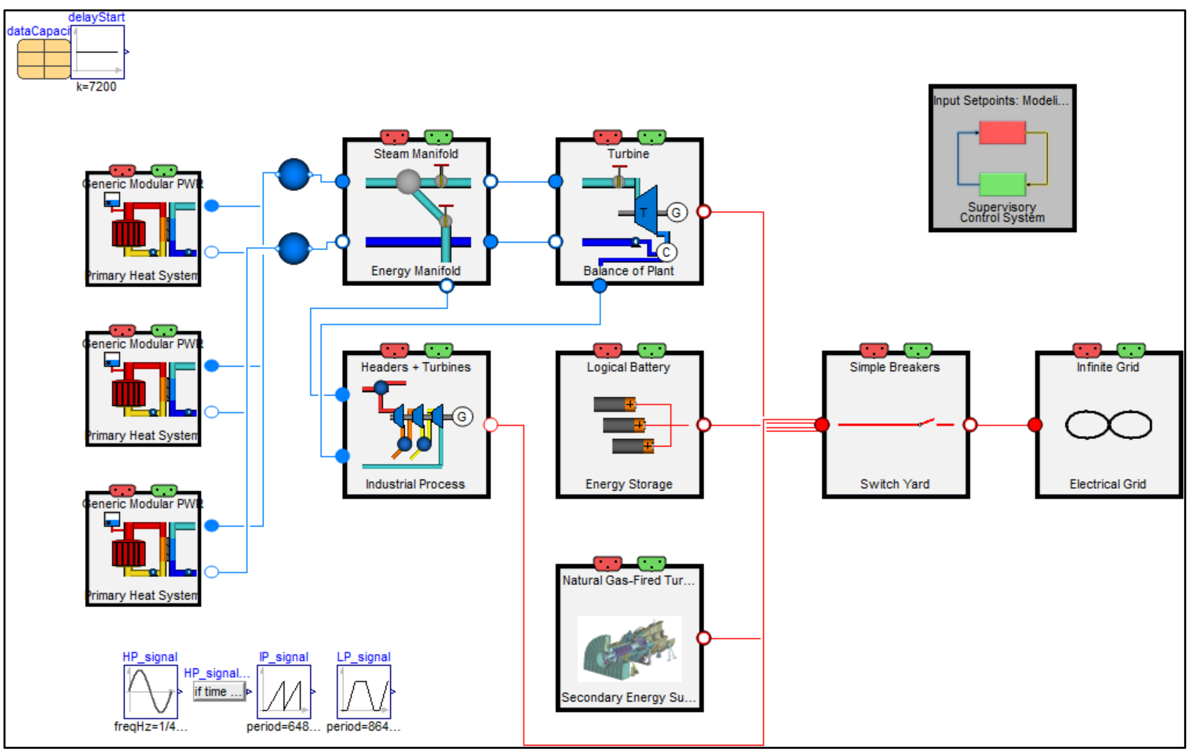

Figure 5. Modelica model of three nuclear reactors connected in parallel in an NHES.

3.2 RESULTS AND DISCUSSION

Each of the scenarios were simulated over a 24-hour time interval, with data being recorded every 10 seconds. This time step is sufficiently frequent for the transient behaviors investigated in the work [25]. The driving signals sent to the model included mass flow rates removed from the IP steam headers and the electrical demand from the BOP, energy manifold (EM), and secondary energy supply (SES). Figure 6 and Figure 7 present these driving signals. Each simulation was run using Dymola 2019 with the Esdirk45-a solver and Visual Studio 2012 compiler. Note that many of the results for certain scenarios and parameters overlap for Figure 9 and beyond and therefore although the legend lists multiple items, i.e., PHS A, B, and C, only one or two lines may be visible as the behavior of items are identical, thus their plots overlap. Additionally, the plots are intentionally small to facilitate general behavior comparison as the quantifiable behavior is not important in this report. Larger images, as indicated with *, are available in the appendix.

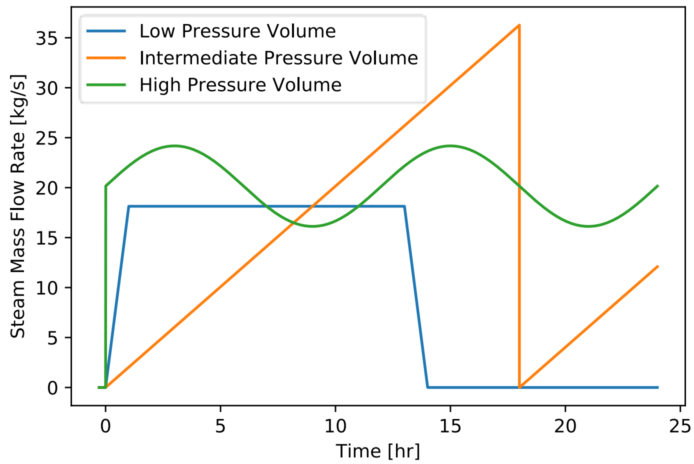

Figure 6. Steam mass flow demand signal for removing steam from the specified steam header of the IP.

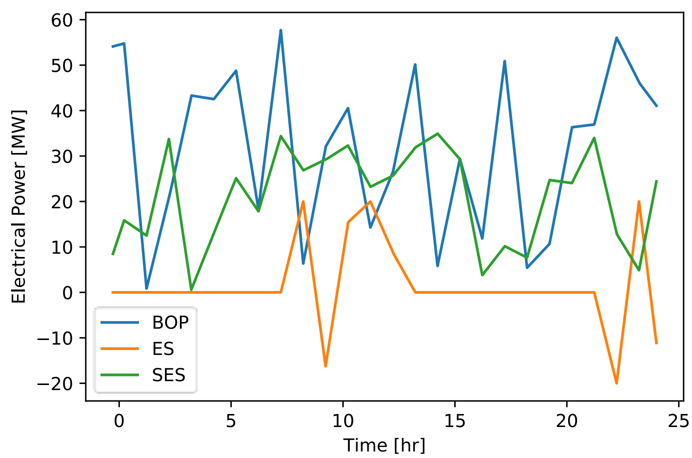

Figure 7. Electrical power demand setpoint signals from the supervisory control system.

3.2.1 Central Processing Unit (CPU) Time

The central processing unit (CPU) time required for each scenario varied. The comparison is shown in Figure 8, where time before zero time represents initialization and a brief period to allow initialization transients to die out. The figure illustrates the impact that various modeling implementation can make. For these results, allowing load following of one of the reactors significantly impacted the simulation time. Optimized/alternative control strategies may be able to reduce this time penalty.

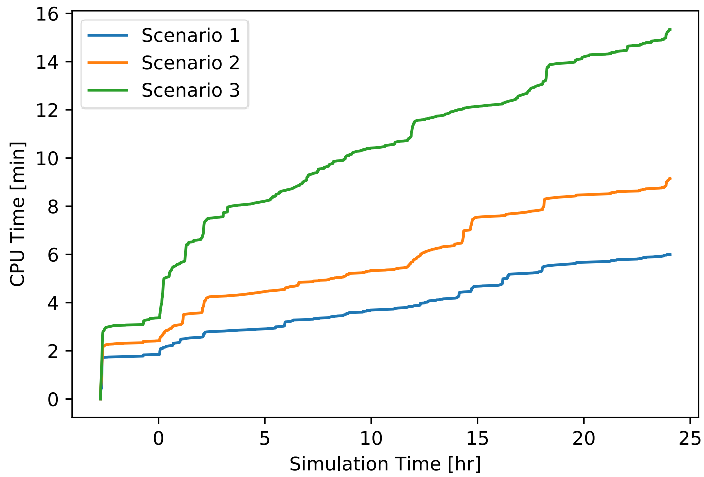

Figure 8. CPU time vs. simulation time for each simulated scenario.

3.2.2 Primary Heat System Response

Figure 9 through Figure 14 present a variety of variables from three nuclear reactors or PHSs. These plots demonstrate that when the three systems have identical control systems, their responses are identical. Thus, they are truly in parallel, as they should be. The steady-state control system (Scenario 2) significantly limits the range and amplitude of any dynamic behavior occurring in the plant, effectively isolating the plant from any oscillations in the balance on the NHES. The load-following behavior and its impact on the plant behavior (Scenario 3) is evident upon consideration of the presented plants. One concern with the implemented load-following control scheme is that, since its measurement signal is power rather than temperature, there are significant fluctuations in the reactor core effective, i.e., mass-averaged, temperatures (Figure 10 and Figure 11). An improved control system would be better suited to key on an absolute temperature (e.g., core outlet temperature), working in tandem with the steam generator flow rate. Properly tuned, this would also likely reduce the fluctuations in the steam generator quality (Figure 15), thereby producing a system with less thermal stresses and likely more stable systems states (e.g., steam pressure).

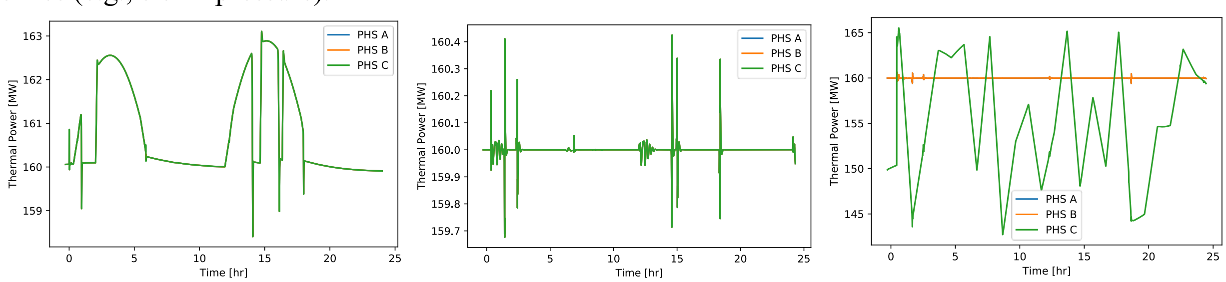

Figure 9. Reactor thermal power output (left to right: scenarios 1–3).

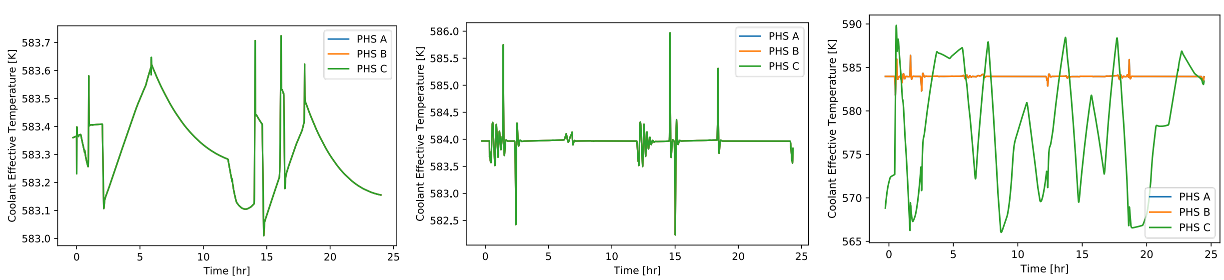

Figure 10. Reactor core coolant effective temperature (left to right: scenarios 1–3).

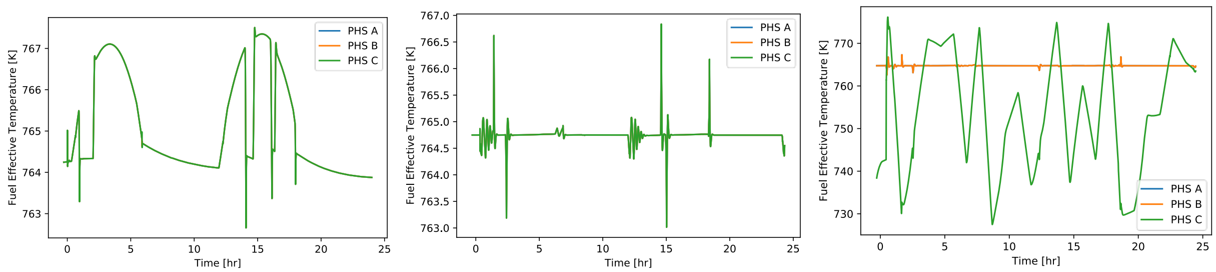

Figure 11. Reactor core fuel effective temperature (left to right: scenarios 1–3).

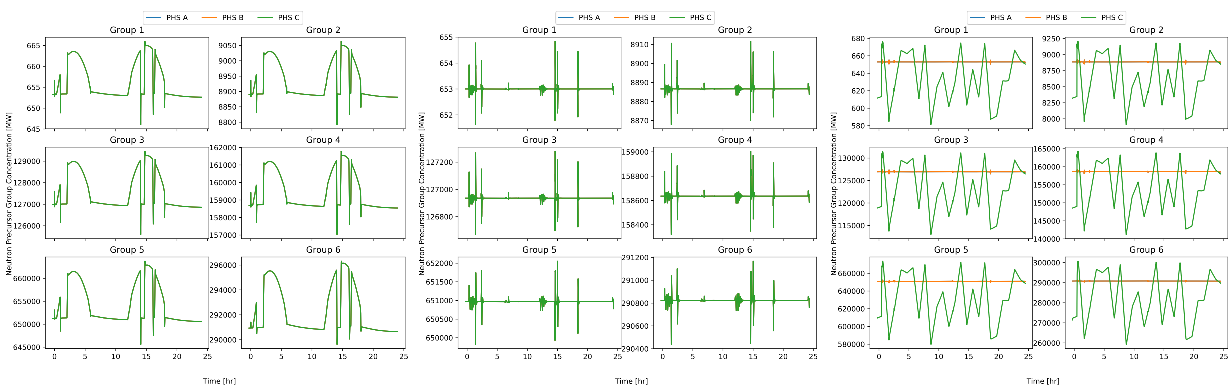

Figure 12. Neutron precursor power-based concentration (left to right: scenarios 1–3).*

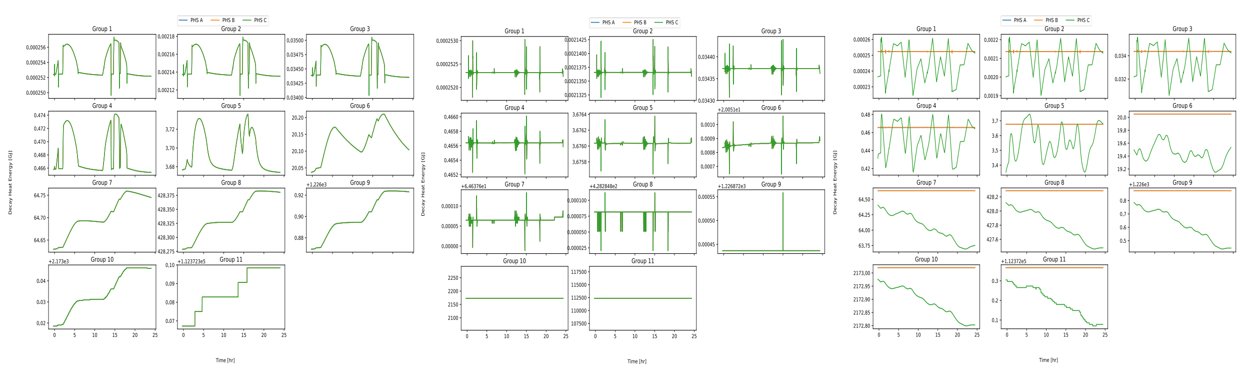

Figure 13. Decay heat group energy-based concentration (left to right: scenarios 1–3).*

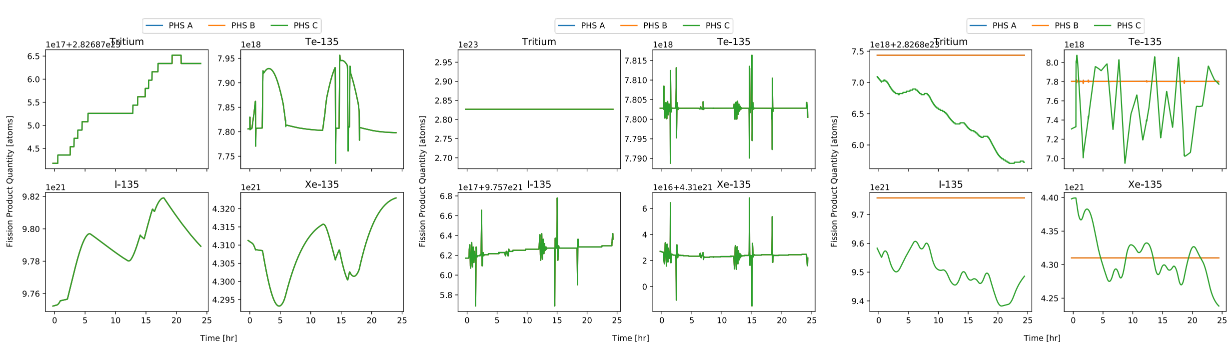

Figure 14. Fission product behavior (left to right: scenarios 1–3).*

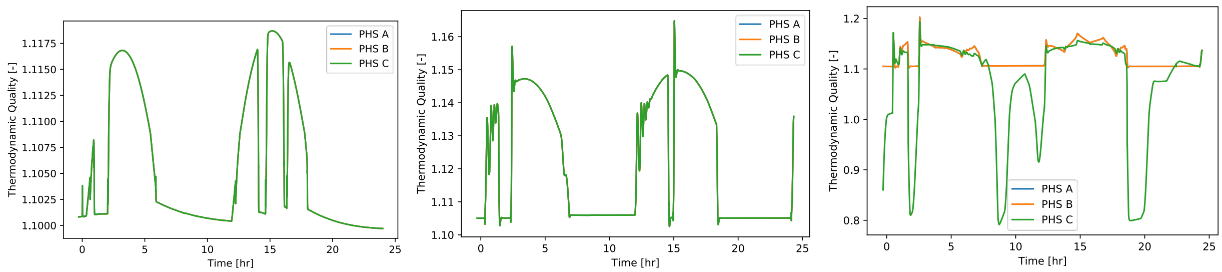

Figure 15. Steam generator outlet quality (left to right: scenarios 1–3).

3.2.3 Electrical Power Generation

Figure 16 through Figure 18 present the principal variables which determine the electrical power generated from each of the steam turbines. Once again, the control system scenario modifies the behavior of the system, requiring more dynamic behavior for a load-following maneuver. Given the use of an ideal pump which fixes the HP steam header in the IP constant at the nominal operating pressure and an ideal temperature boundary that ensures the superheated steam condition, many dynamics will likely be suppressed. Removing these assumptions as the model matures will be very important to avoid dampening or completely suppressing realistic behaviors. The IP consumption of Figure 18 represents the energy required to keep the temperature at the required setpoint. The reason this is so large compared to the overall power production stems from the nominal operating conditions employed. By default, the PHS generates steam at a lower pressure and temperature than the nominal HP steam header. Therefore, given the large mass flow rate of steam, there is a corresponding large heating requirement. As stated, removing this assumption will greatly improve the realism of the model.

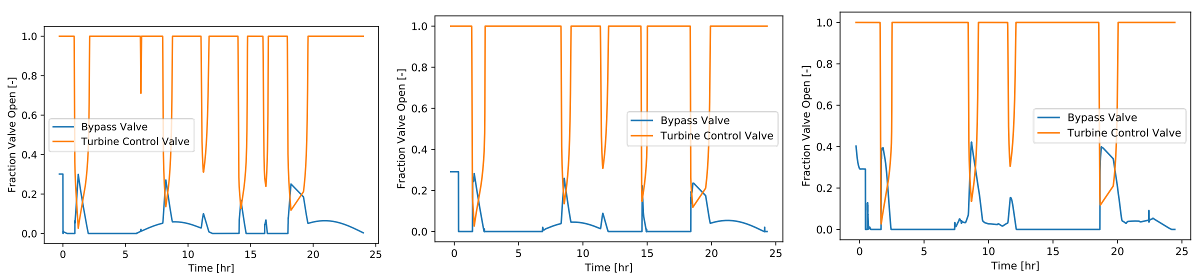

Figure 16. BOP turbine control and bypass valve position (left to right: scenarios 1–3).

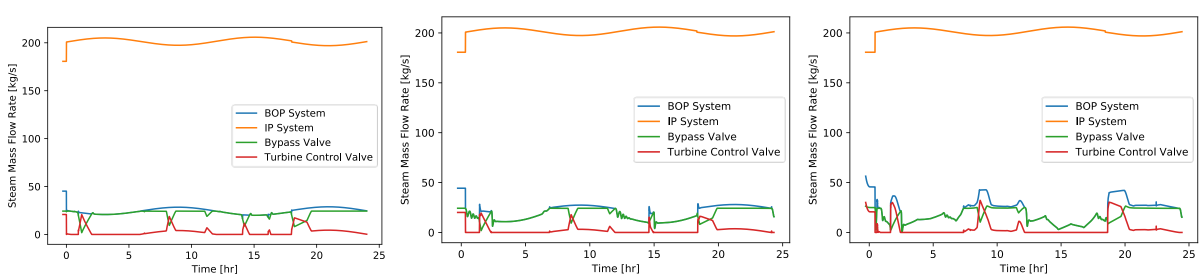

Figure 17. Mass flow rates (left to right: scenarios 1–3).

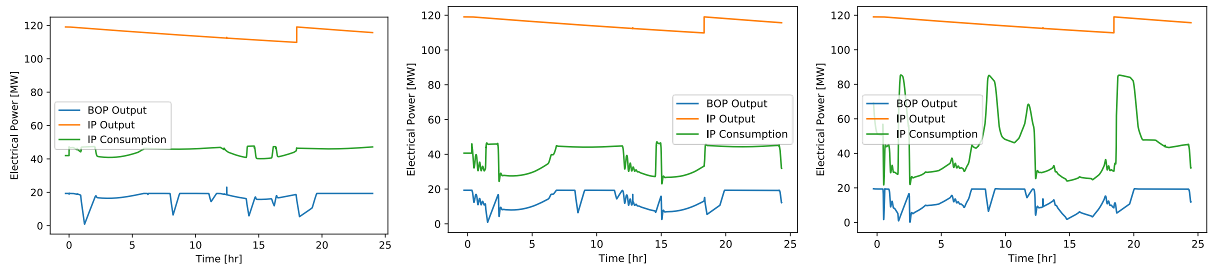

Figure 18. Electric power generation/consumption (left to right: scenarios 1–3).

3.3 RELIABILITY

3.3.1 Component Reliability Model Results

The 24-hour Modelica run results were fed into the component model to compare two cases to understand load-following operation effect on TCV. In Figure 19 and Figure 20, the dot distributions at the regression line graphs presents the TCV's health for the Scenario 2 and 3, the difference between the steady state and with load-following operation of the one reactor has been listed in Table 8.

The estimated η value for Scenario 3 decreased negligibly (19 hours over the component characteristic life), representing an accelerated deterioration process with load-following. This is important and expected. The characteristic lifetime of the component under both operations is calculated as 7.86 years. These preliminary results will be updated or verified as more detailed physical systems are incorporated into the model and more representative load-following operations are defined.

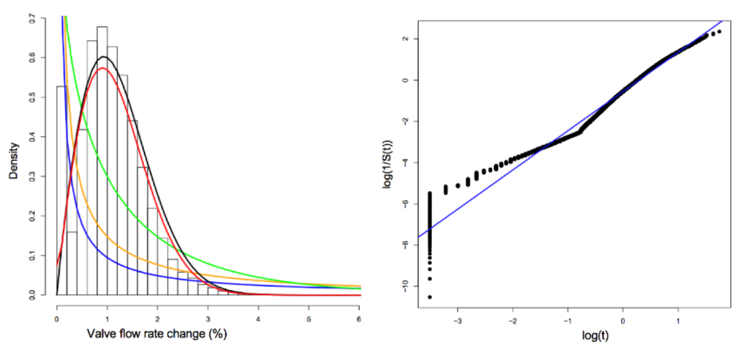

Figure 19. Weibull analysis results and fitting statistics for TCV for 24-hour run for Scenario 2

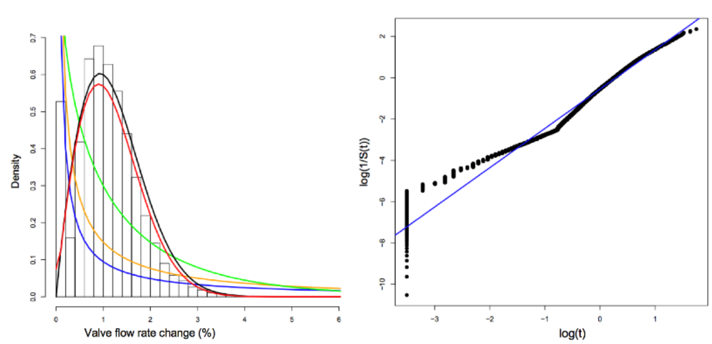

Figure 20. Weibull analysis results and fitting statistics for TCV for 24-hour run for Scenario 3

Table 8. Weibull parameters and failure rate estimations with one-hour Bayesian updates

Scenario	β	η [hours]	*Failure Rate* λ	E[λ|z] with Uniform (β = 1.35, 1.4)
2	1.383	68,851	1.452E-05	1.652E-05
3	1.383	68,832	1.453E-05	1.652E-05

4. SUMMARY

The principle task of this report is to show the feasibility of coupling of the reliability and Modelica models and demonstrate multiple reactors operating in parallel which can be applied to further work. This flow has been successfully demonstrated. For this purpose, a generic modular nuclear reactor with a nominal rating of 160 MWt has been created and included within the NHES Gitlab repository. This model was generated based on open literature and by simplifying assumptions to capture the general dynamics of a nuclear reactor, complete with kinetic behavior, decay heat, and fission product feedback. This model has been simulated in a multimodular system consisting of three reactors operating in parallel to provide steam to a steam header and a step-down turbine industrial process inspired by the CHP facility operated by Eastman in Kingsport, Tennessee. The simulations included in this report cover three operating conditions for the nuclear reactors: (1) no direct control, (2) steady-state power control, and (3) two reactors operating at steady-state with one allowed to load follow. Under all scenarios, time varying demands were applied on the system to generate dynamic feedback which propagated through the model. This report demonstrates the feasibility of modeling multimodular systems and will be expanded to include improved control systems which will likely decrease stresses on the system and will provide a more realistic steam header/turbine system that is similar in complexity to the Eastman CHP.

The component reliability model has been tested for the two scenarios described; steady-state power control, and two reactors operating at steady-state and one with load follow to investigate load-following effect on a TCV reliability. It has been shown that characteristic life time of the component decreased 19 hours under the load-following but failure rate results for 24-hour runs are almost identical with the current state of the component model. Future work exploring more realistic physical models and advanced control systems may significantly change the component performance and it will therefore be important to reanalyze the system and component performance. The other ongoing effort is on defining the multiunit dependencies and creating the subsystem time-dependent reliability model for the PHS to capture the inter-unit dynamics with other subsystems.

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APPENDIX A

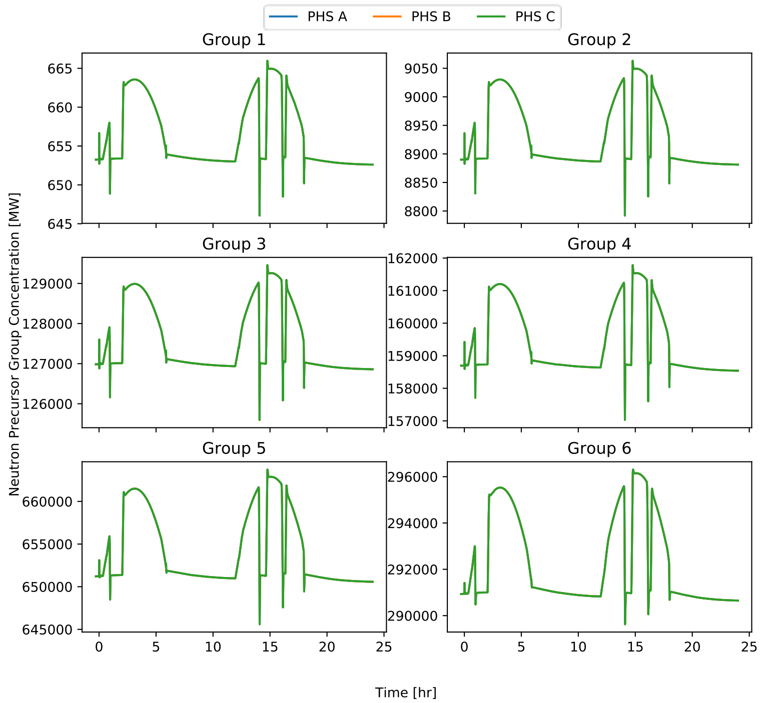

Figure A 1 Neutron precursor power-based concentration Scenario 1

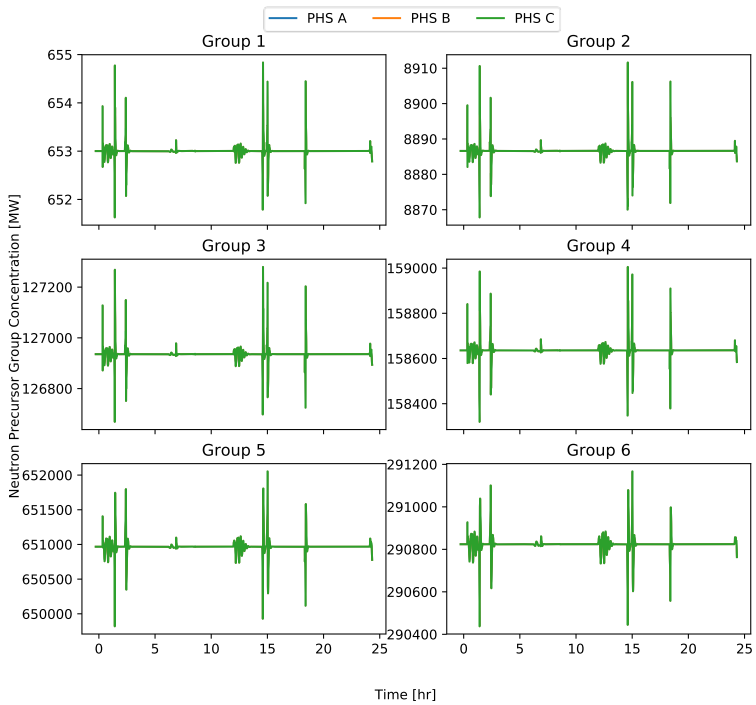

Figure A 2 Neutron precursor power-based concentration Scenario 2

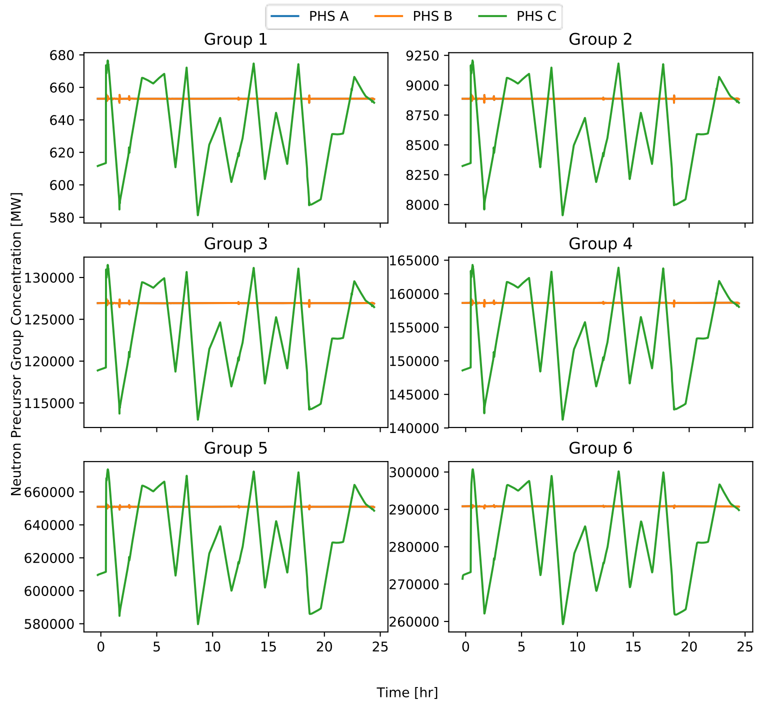

Figure A 3 Neutron precursor power-based concentration Scenario 3

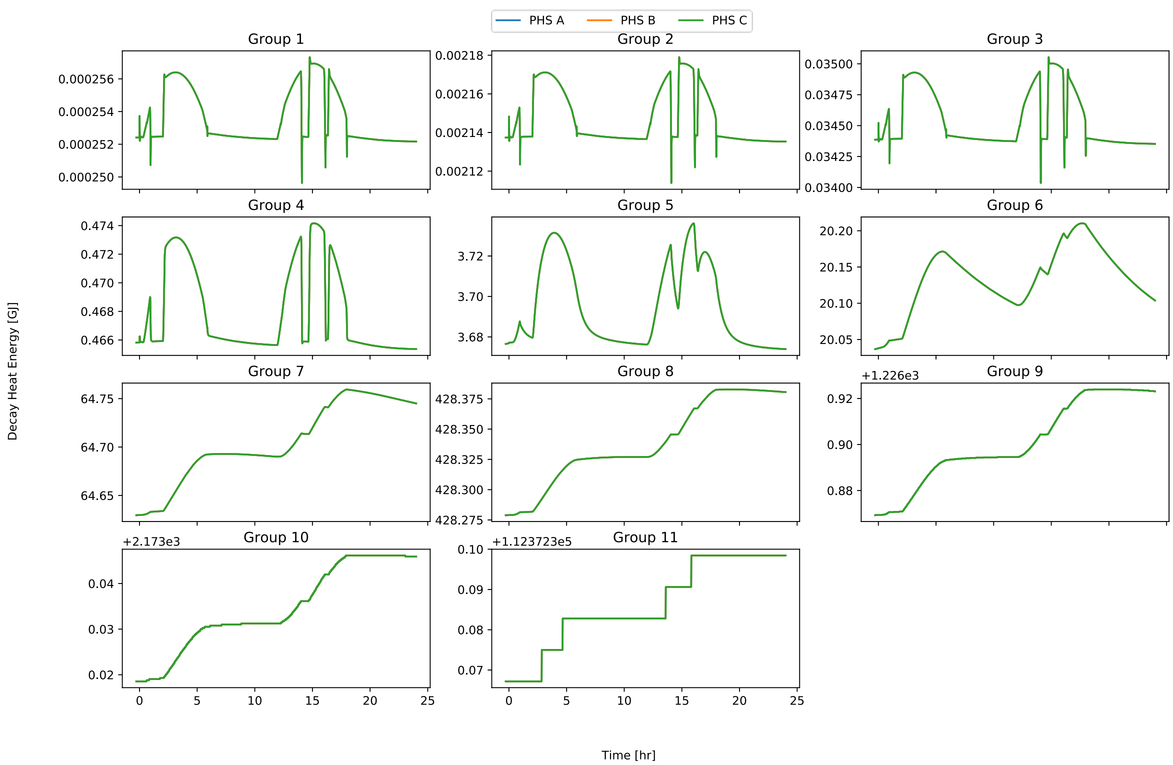

Figure A 4 Decay heat group energy-based concentration Scenario 1

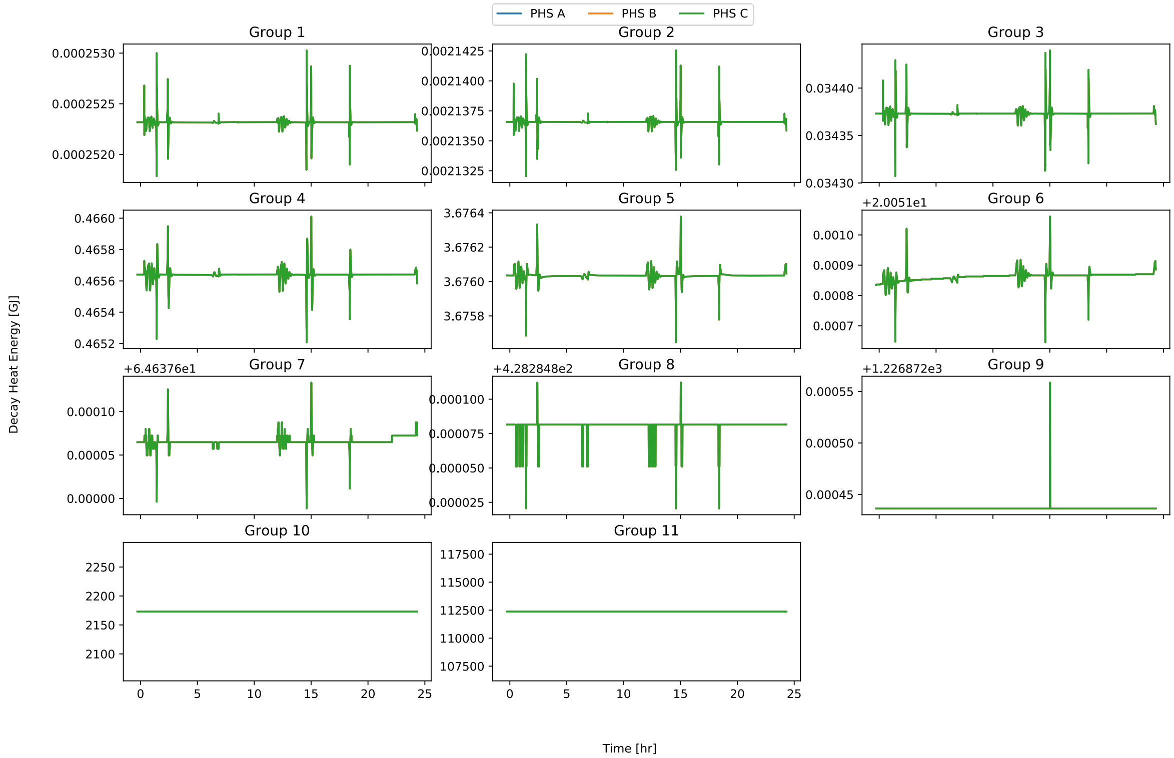

Figure A 5 Decay heat group energy-based concentration Scenario 2

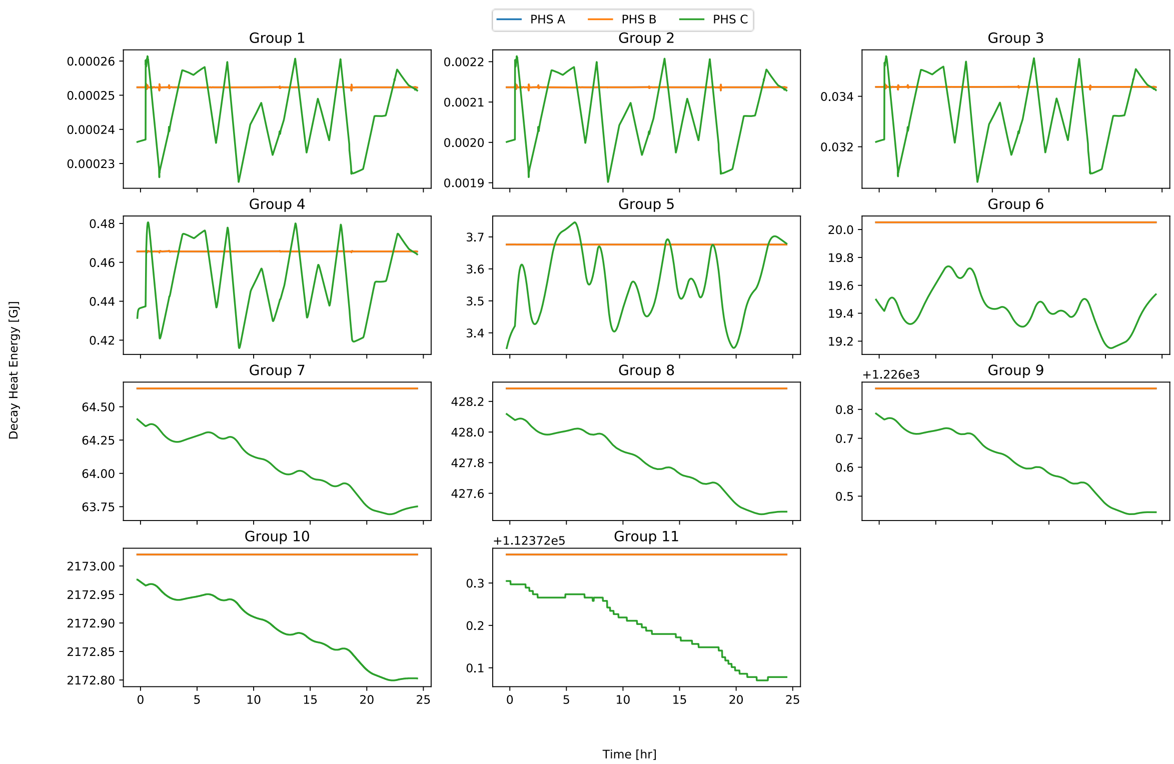

Figure A 6 Decay heat group energy-based concentration Scenario 3

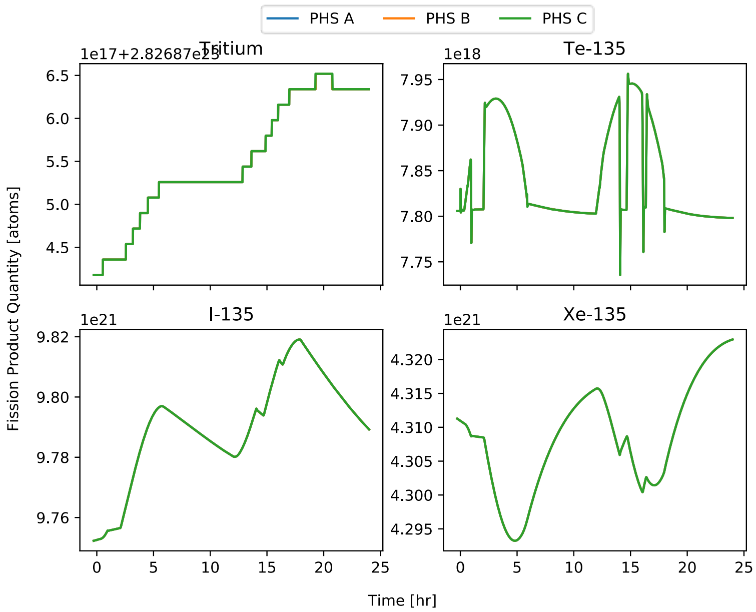

Figure A 7 Fission product behavior Scenario 1

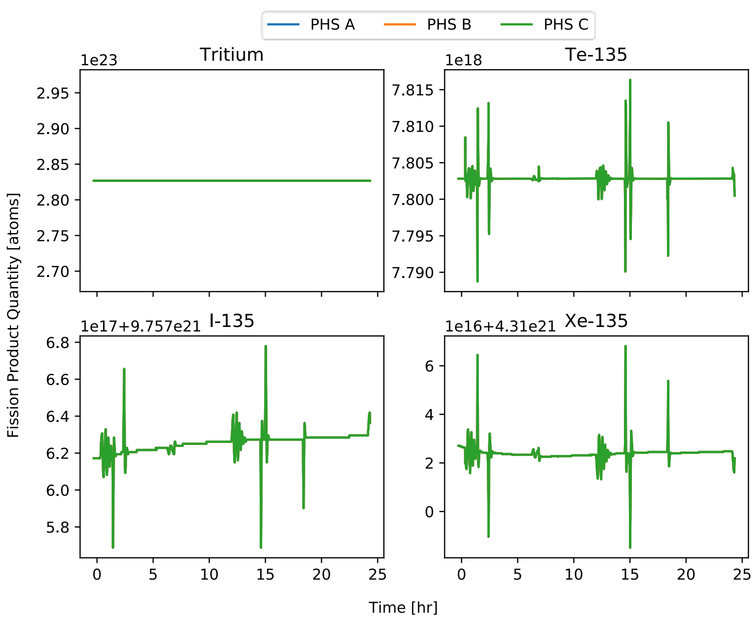

Figure A 8 Fission product behavior Scenario 2

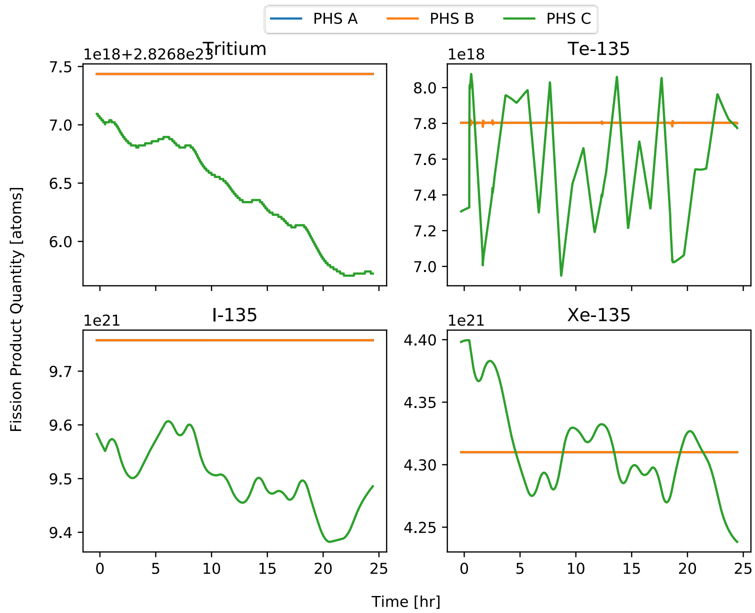

Figure A 9 Fission product behavior Scenario 3