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Simulation of Small Modular Reactor under Specified Transients

MAGY M. KANDIL
Nuclear and Radiological Regulatory Authority (ENRRA)
Cairo, Egypt

Abstract

In this research, a new dynamic model of a small modular reactor is developed. The developed modeling consistent of a lumped parameter of fuel to coolant heat transfer combines with the vessel upper and lower plenums coolant masses for representing the power of core and the response of heat transfer dynamics by simulating the core plant operation under specified transients. The developed model simulated Babcock & Wilcox Generation mPower as a small modular reactor. The developed model is validated by comparing with the model of reference [1]. The results show that, the developed model has the same trend. In addition, the developed model simulates the reactor behavior under specified transients of perturbations of control rod withdrawal and increases in inlet core temperature. The analysis of the results shows that the developed model has a capable of predicting the dynamics behavior of the reactor under the specified transients.

Keywords: Nuclear Power Plant, Small Modular Reactor, Babcock & Wilcox Generation mPower reactor, MATLAB/Simulink.

1. INTRODUCTION

Small modular reactors (SMRs) are growing in attention through the world Starting in the last decade. Compared with the today's large pressurized water reactors (PWRs), SMRs have distinguishing features. Among many others, the most protuberant differences can be brief as: (1) many SMRs have an integral design that utilizes both pressurizer and steam generator inside the reactor pressure vessel, and (2) these new designs are essentially safer and more secure by insertion the SMR into a pool under ground level, rejecting primary coolant pumps and, consequently, associated failure modes [2].

In previous many researches are made for nuclear plants dynamic modeling such as Arda et al. [3,4] evaluated a nonlinear dynamic modeling for NuScale reactor. Their implemented model is a grouping of neutronics and thermal-hydraulics for the reactor core, with a single tube illustration through time-varying boundaries and three regions which are subcooled, boiling, and superheat water inside a helical coil once-through steam generator (SG) [1]. Due to the existence a typical pressurizer where heater and spray are installed inside it, the model is developed based on the conservation of fluid mass, volume, and energy. Available researches run on SMR type of reactors have focused on density wave oscillation analysis based on the drift flux model, which can frequently be applied in boiling water reactors (BWRs) [5]. SMR reactor types are categorized as something between pressurized water reactors and boiling water reactors.

In NPPs, typical dynamic models for predicting their systems behavior during the transient situations are essential. Thermalhydraulic codes such as ATHLET simulate the behavior of the thermalhydraulic systems through numerical methods [6], but separated nonlinear differential equations for safety system analysis cannot be generated such as stability analysis of controller design analysis. Appropriate functional model is required to establish the base of controller design according to one of the control models, such as transfer function, state space, direct Lyapunov theory, etc.

In this research, anew dynamic model for SMR is developed to analyze the response of the SMR under specified transients (disturbances). The developed model is applied analytically at mPower SMR, to represent its essential dynamics behavior under specified transients which evaluated within the MATLAB/Simulink dynamic environment. The physical reactor model is described in Section 2; developing the new dynamic model is represented in Section 3; Section 4 represented the methodology of the developed model simulates mpower reactor, and the validation of the developed represented in section 5. The results of the developed model applied under specified transients are described and dissection in section 6 and conclusion is presented in Sections 7.

2. REACTOR OVERVIEW

The mPower plant consists of an integral PWR small modular reactor and related balance of plant, designed by Generation mPower LLC to generate a nominal output of 195 MW(e) per module as shown in fig. 1 the Babcock & Wilcox (B & W) Company [1]. The standard plant design, for each mPower plant contains a 'twin-pack' set, or two mPower reactor modules, generating a nominal 390MW(e).

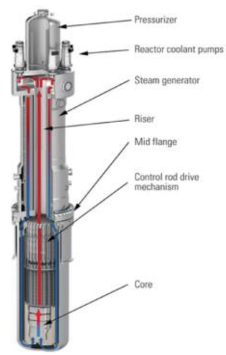


Fig. 1: B&W Generation of mPower Reactor Pressure Vessel [1]

3. THE DEVELOPED MODEL

The developed model is considered as typical analysis of a thermal fluids system involves conservation of mass, momentum and energy assuming that the reactor coolant is at constant density, pressure and mass flow rate. The energy balance accounts for the heat transfer from the nuclear fuel to the coolant using Mann's model [7].

The core calculations have been done by the developed model which has abilities to calculate the neutronic and thermal hydraulic parameters and the hot leg riser and cold leg region as upper and lower plenums. The developed model uses a combination method for simulation reactor core to predict its output parameters under specified transient desired core behavior. Needed parameters for training are extracted from the model calculations. The dynamic model capable of predicting system behavior in normal condition or under specified transient condition with reactivates feedbacks.

The developed dynamic model is appropriated specified transient conditions, which are controlled only by the control rods.

The input variable for the developed model is the external reactivity and changes of inlet temperature from lower plenum.

The developed model considered as, a software package using Ordinary Differential Equations (ODEs) and Simulink model which that are projected to predict the reactor behavior in transient states. In the developed model ODEs consists of a combination of neutronics and thermal-hydraulics equations.

3.1. Reactor neutrons

In the developed model neutronics equations are created with point kinetic theory, with six precursor groups [8]. The reactor core sometimes divided into many nodes [2]. In developed model, the fuel mass remains as a single node.

In this research, the physics of the reactor core are modeled through six neutron precursor concentrations and the total reactivity by point kinetics equation that are expressed in terms of reactor thermal power (P) through the balance equations as:

$$\frac{dP}{dt} = \frac{\rho_t - \beta}{\Lambda} P + \sum_i^6 \lambda_i C_i \tag{1}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} P - \lambda_i C_i \tag{2}$$

Where: t is the time (sec), P is the fractional reactor power, ρ_t is the total reactivity of the system, β is the sum of delayed neutron fractions, Λ is the neutron mean lifetime, λ_i is the i:th decay constant for the delayed neutron precursor, C_i is the concentration of the i:th fraction of the delayed neutrons' precursors and β_i is the i:th delayed neutron fraction.

3.2. Feedback Coefficients

SMRs can be operated at different power levels through changes in the position of control rods that considered as an external reactivity and controls by reactivity feedback due to changes in fuel and coolant temperatures donate to the total system reactivity that expressed In ODE as:

$$\partial\rho_t = \delta\rho_{ext} + \alpha_F \delta T_F + \alpha_C \delta T_C + \alpha_P \delta T_P \tag{3}$$

Where α_F is the reactivity feedback coefficients of fuel, α_C is the reactivity feedback coefficients of coolant temperature and α_P the reactivity feedback coefficients of primary coolant pressure, δT represent the deviation from the steady-state for fuel (F) and coolant

(C) temperatures, δp is the deviation from the steady-state for primary coolant pressure (p) and $\delta_{\rho ext}$ is the reactivity induced by control rod movement.

In the design of a nuclear reactor, temperature feedback coefficients play an important role as safety system since they control reactivity of the reactor to avoid reaching a prompt supercritical state [9]. These coefficients are negative means that as the temperature of the fuel and coolant increases, the total reactivity of the system decreases and causes the core system to return to its nominal safe temperature.

3.3. Reactor thermal-hydraulics

In the developed model, a thermal fluids system typically analyses depended on the conservation of momentum, mass and energy. Simplification of the analysis is doing by assuming that the reactor coolant has constant density, mass flow rate and pressure; due that the state equations for momentum and mass are eliminated. The energy balance accounts for the heat transfer from the nuclear fuel to the coolant using Mann's model as shown in fig. 2 [7]. In this model, the temperature difference is taken as the difference between the fuel temperature and the average temperature of the first coolant lump to provide better physical illustrations.

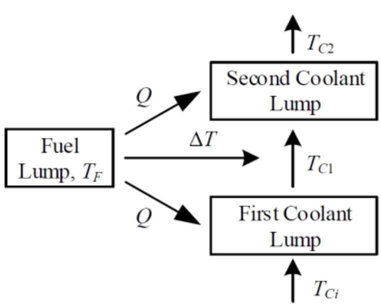


Fig. 2. Mann's heat transfer model

4. THE DEVELOPED MODEL SIMULATED THE MPOWER SMR

The Developed model simulated mpower with MATLAB/Simulink as shown in fig. 3 and through system of differential equations that solved to predict the behavior of the reactor in specified transients after linearized them as shown in the following sections.

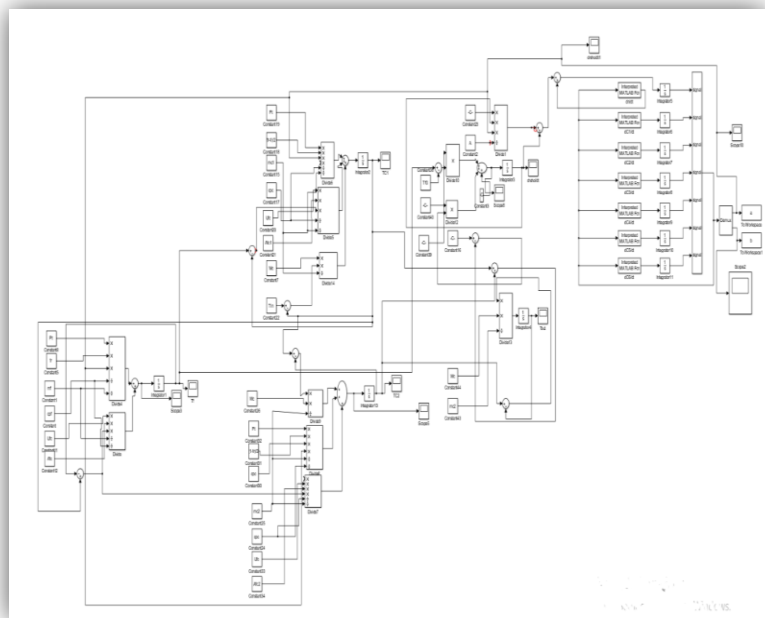


Fig. 3. The developed model with Simulink

4.1. Reactor neutronics differential equations

In the developed model the change of total reactivity is determined using (differential Equation 4), the effects of the fuel and coolant temperature coefficients and the external reactivity (ρ_{ext}) are considered. Due to pervious assuming that the primary fluid has constant pressure, so that $\alpha_P \delta T_P$ is removed and others terms are linarites as:

$$\frac{d\,\partial\rho_t}{dt} = \alpha_F\,\partial T_F + \frac{\alpha_C}{2\Lambda}\partial T_{C1} + \frac{\alpha_C}{2\Lambda}\partial T_{C2} + \partial\rho_{ext} \qquad (4)$$

The physics of the reactor core are modeled with constant parameters of the diffusion equation of six delayed neutrons precursor concentrations are shown in table 1 and the fuel and coolant temperature feedback coefficients are shown in table 2.

Table 1: The decay constants and the delayed neutrons fractions due to the thermal fission of uranium-235

Group	λi , [s-1]	Bi
1	0.0124	0.000221
2	0.0305	0.001467
3	0.111	0.001313
4	0.301	0.002647
5	1.14	0.000771
6	3.01	0.000281
Sum		$\beta=\sum_i^6 \beta i$ =0.00067

After replacing the reactivity term (ρ_t), in Equation (1) with its equivalent, the point kinetics equations 1&2 linearized as following:

$$\frac{d}{dt}\partial P = \frac{-\beta}{\Lambda}\partial P + \sum_{i=1}^{6}\lambda_i \partial C_i + \frac{\alpha_F}{\Lambda}\partial T_F + \frac{\alpha_C}{\Lambda}\partial T_C + \frac{1}{\Lambda}\partial \rho_{ext} \quad (5)$$

$$\frac{d}{dt}\partial C_i = \frac{\beta}{\Lambda}\partial P - \lambda_i \partial C_i \quad (6)$$

4.2. Reactor thermal-hydraulics differential equations

The linearized equations of the reactor thermal hydraulics are given by equation from (7) to (9) are being used with their parameters value in table. 2. [1] as shown:

$$\frac{d}{dt}(\partial T_f) = \left(\frac{f P_0}{m_f c_{pf}}\right)\partial P + \frac{U_{fc}A_{fc}}{m_f c_{pf}}(\partial T_{c1} - \partial T_f) \quad (7)$$

$$\frac{d}{dt}(\partial T_{C1}) = \left(\frac{(1-f)}{2}\right)\left(\frac{P_0}{m_{c1}c_{pc}}\right)\partial P + \frac{U_{fc}A_{fc1}}{m_{c1}c_{pc}}(\partial T_f - \partial T_{C1}) + \frac{W_c}{m_{c1}}(\partial T_{lP} - \partial T_{c1}) \quad (8)$$

$$\frac{d}{dt}(\partial T_{C2}) = \left(\frac{(1-f)}{2}\right)\left(\frac{P_0}{m_{c2}c_{pc}}\right)\partial P + \frac{U_{fc}A_{fc2}}{m_{c2}c_{pc}}(\partial T_f - \partial T_{C1}) + \frac{W_c}{m_{c2}}(\partial T_{c2} - \partial T_{C1}) \quad (9)$$

Where: T_f is the average temperatures of the fuel, T_{c1}, and T_{c2} are first and second coolant lumps, respectively, while T_{lp} is the lower primary temperature at the steam generator outlet. c_{pc} is the coolant heat capacity, c_{pf} is the fuel heat capacity, m_c is the total mass of coolant in core, m_{c1} is the coolant mass of node1; m_{c2} coolant mass node2, f is the fraction of the total power, U_{fc} are the heat transfer coefficient from fuel to coolant, A_{fc} is the effective heat transfer surface area, m_f mass of fuel, P_0 nominal power, and finally Wc primary coolant mass flow rate.

4.3. The differential equations of the Hot leg riser and cold leg region in upper and lower plenums

The hot leg riser and cold leg region in upper and lower plenums are linearities to produce differential equations as flowing:

$$\frac{d}{dt}(\partial T_{HL}) = \frac{W_c(\partial T_{C2} - \partial T_{HL})}{W_{HL}} \quad (10)$$

$$\frac{d}{dt}(\partial T_{CL}) = \frac{W_c(\partial T_{lP} - \partial T_{CL})}{W_{CL}} \quad (11)$$

Where: $T_{HL}$and T_{CL} represented the average temperature of the hot leg riser and cold leg regions respectively, m_{HL} and m_{DR} are the coolant mass of the hot leg riser and cold leg regions respectively, and W_{HL} and W_{CL} are the mass flow rate inside the particular region, hot leg riser and cold leg regions.

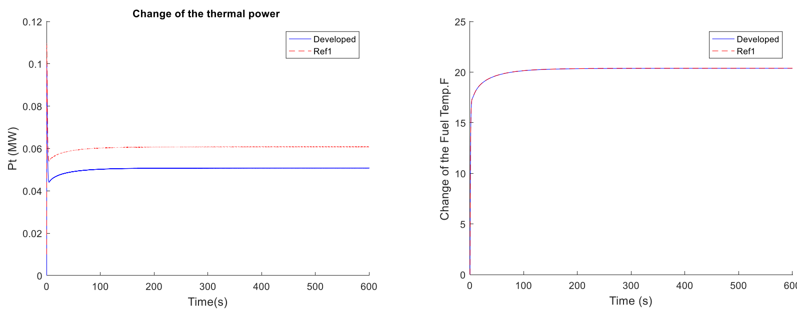
Table .2. Parameters of the thermal-hydraulics differential equations

Parameter	Value	Unit
A_{fc}	14120.6	$[ft^2]$
A_{fc1}	7060.3	$[ft^2]$
A_{fc2}	7060.3	$[ft^2]$
c_{pc}	1.376	$[\frac{BTU}{\text{lbm \_F}}]$
c_{pf}	0.059	$[\frac{BTU}{\text{lbm \_F}}]$
f	0.97	$[-]$
m_c	6898.3	$[lbm]$
m_{c1}	3449.1	$[lbm]$
m_{c2}	3449.1	$[lbm]$
m_f	53850.5	$[lbm]$
P_0	502343	[BTUs]
T_f	962.672	[F]
T_{f0}	962.741	[F]
U_{fc}	0.0909	$[\frac{BTUs}{ft^2F}]$
Wc	8333.3	$[\frac{\text{lbm}}{\text{S}}]$
α_C	-0.0002	$[\frac{1}{\text{F}}]$
α_F	-0.165e-05	$[\frac{1}{\text{F}}]$
β	0.0067	[-]

5. THE DEVELOPED MODEL VALIDATION

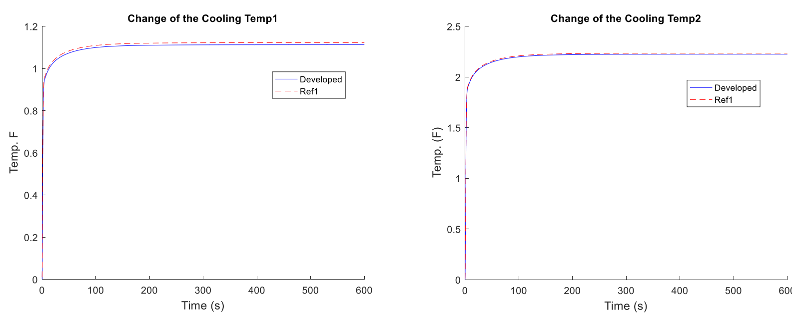
The validation of the developed model is required to ensure that it can determine and predict the response of the reactor core under specified transients thus, a comparison of reactor core dynamic simulations between the developed model results and previously published results in reference [1] is applied. The comparison transient simulates the system of the two model's response to 10 cents (¢) step insertion as positive step reactivity (e.g., direct control rod withdrawal). Supposes that, the two reactors are operating at steady state at time t = 0, the reactivity insertion occurs.

The comparison between changes of the thermal power and other core output variable in reference [1] and the developed model are shown in fig. 3& fig. 4. In both models the transient change in power and other variables of the reactor core is the same trend of the value of reference [1].



(a) (b)

Fig.4. Change of the thermal power reactor and the fuel temperature with $\partial\rho_{ext}$(+10¢).



(a) (b)

Fig.5. Changes of the first and second coolant temperatures with $\partial\rho_{ext}$(+10 ¢).

From the dynamic reactor performance in the fig. 4 & fig. 5, it shows that the developed model of B&W mpower results are near reference [1] during transient simulations, of an insertion of 10 ¢ rod control reactivity. Moreover, the external control rod reactivity insertion causes the average reactor coolant temperature to increase, creating a mismatch to the average temperature setpoint. The mismatch initiatives the feedback reactivity of the temperature for fuel and cooling that introduce negative reactivity to compensate for the effects of the step insertion. Over time, total reactivity stabilizes and the reactor parameters return to their nominal values with new steady state variables as shown in all figs.

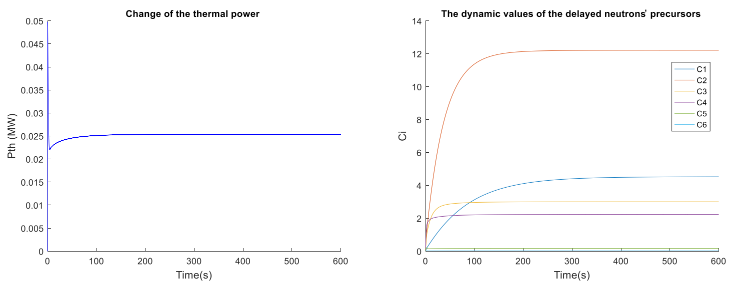
6. DYNAMIC PERFORMANCE OF THE DEVELOPED MODEL UNDER SPECIFIED TRANSIENTS

For verified the developed model, specified transients are applied to analysis the dynamic performance of the core reactor by a step increases in two input variables, the external reactivity (ρ_{ext}) that results from the control rod position and feedwater temperature for the core (lower primary coolant temperature (T_{lP}). The specified transients are applied independently at (0 sec) of the steady-state operation to demonstrate that the simulation starts from a stable point and so that the initial steady-state conditions can be seen visibly. Figs. 6–13 display responses of the different nuclear output variables for the abovementioned perturbations. For each case, the input variable under investigation is perturbed from its initial steady-state value.

6.1. Response of the step change in external reactivity

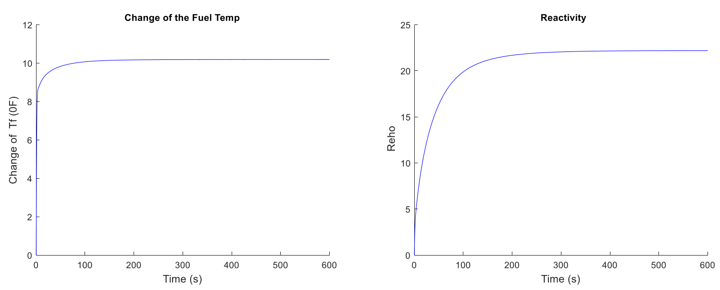
The first specified transient is considered, as the external reactivity (ρ_{ext}) increases due to control rod withdrawal with 5 ¢ at 0s causes increasing of the fission rate and neutron flux and, correspondingly, a prompt jump in reactor thermal power (P) and in the dynamic values of the delayed neutrons' precursors, as shown in fig. 6a & fig.6b respectively. Following the increased thermal power generation, the fuel temperature rises and the total reactivity too as shown in fig. 7a &fig. 7b and more heat is transferred from the fuel region to the primary coolant in the core that increases the coolant temperature node 1 (T_{c1}) and node 2 (T_{c2}) as shown in fig. 8a & fig. 8b. This increase in the core exit coolant temperature propagates through the hot leg riser to the steam generator inlet temperature as shown in fig. 9a, but the cold leg zero change shown in fig. 9b.

The Results from the fig. 6 & fig. 7 show also that, the increases in the total reactivities with $\partial\rho_{ext}$ (+5¢) act causes an increase in the fission rate and neutron flux, congruently, an initial prompt jump in the fractional reactor thermal power. Subsequent the fuel temperature increases, as the fuel temperatures increases not quickly, the Doppler effect provides a negative reactivity and decreases the power change. In addition, the coolant temperature increases due to fuel heat-up, the coolant temperatures change starts making additional changes in reactivity feedback. The reactivity feedbacks bring a steady-state stable power level (δP) of different values that increased with increase the reactivity inserted. Thus, the mpower as SMR PWR is stable under an external reactivity insertion without any control action but has sudden increase that in thermal power of the core can be danger in high values. The new steady-state fuel (T_f) as shown in fig. 8b and also, the two Cooling temperatures T_{c1} & T_{c1} temperatures rise respectively, as shown in fig. 9 then, they reach to steady state about 100 seconds. In addition, the hot leg (T_{HL}) temperature increases due to coolant temperature (T_{c2}) they reach also, to steady state about 100 seconds.



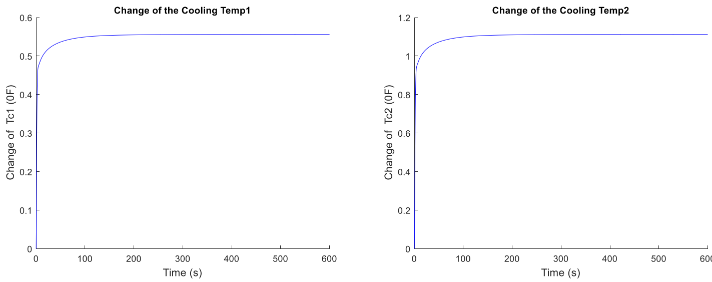
(a) (b)

Fig.6. Change of the thermal power reactor and the dynamic values of the delayed neutrons' precursors with $\partial\rho_{rod}$ (+5¢).



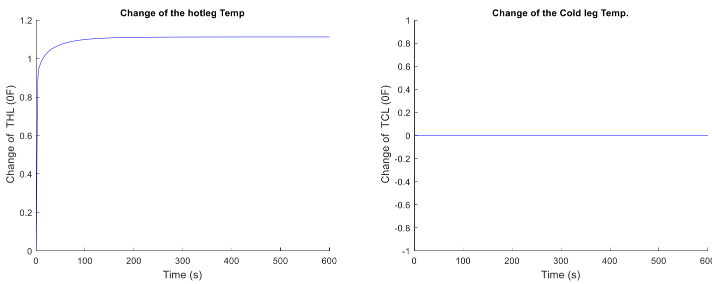
(a) (b)

Fig.7. Change of the fuel temperature and total reactivity with $\partial\rho_{ext}$ (+5¢).



(a) (b)

Fig.8. Changes of the first and second coolant temperatures with $\partial\rho_{ext}$ (+5¢).



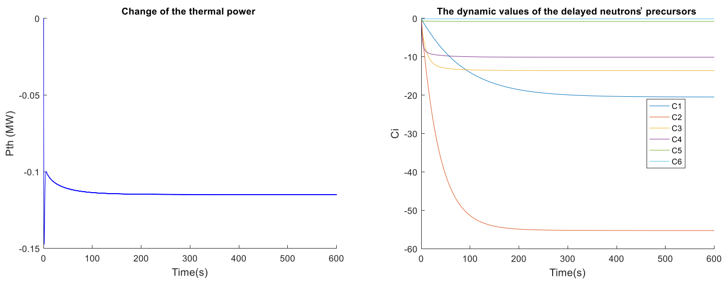
(a) (b)

Fig.9. Changes of the hot leg and cold leg temperatures with $\partial\rho_{rod}$ (+5¢).

6.2. Response to a step change in feedwater inlet temperature

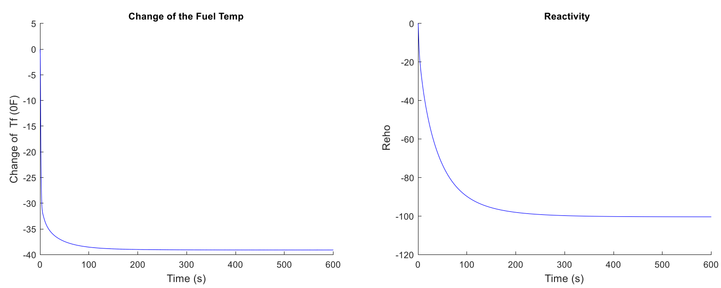
For the second scenario, a 7◦F increases in the inlet feed-water lower primary temperature (∂T_{lP}) that inlet to the core is introduced at t = 0 s. With this perturbation, the required heat for the feedwater to reach the saturation temperature diminishes. The predication outputs from the developed dynamic model are represented in the flowing graphs. In fig. 10a the predicted relative fraction of power is decreased and also in fig. 10b the dynamic output values of the delayed neutrons' precursors decrease. Fig. 11a indicates the characteristics of the fuel is decrease due to decrease in the thermal power and also the change of the total reactivity decreases as shown in fig. 11b. Fig. 12a and fig. 12b characteristic coolant temperatures decrease, as shown. However, the curves clearly demonstrate that the fuel's temperature more decrease than that of the coolants. Fig.13a characteristics of the hot cold outlet temperature decreases but cold leg temperature increases, as shown fig. 13b.

From all figures the output variable of reactor core reach to steady state about 100 seconds due to increase in (∂T_{lP}) inlet temperature, the fuel temperature increases, as the fuel temperatures decrease not quickly, the Doppler effect provides a positive reactivity and increase the power change. In addition, the coolant temperatures decrease due to fuel decrease, the coolant temperatures change starts making additional decrease changes in reactivity feedback thus keep them to steady state. The reactivity feedbacks bring a steady-state stable power level, fuel, coolant and hot leg temperatures as shown in figures.



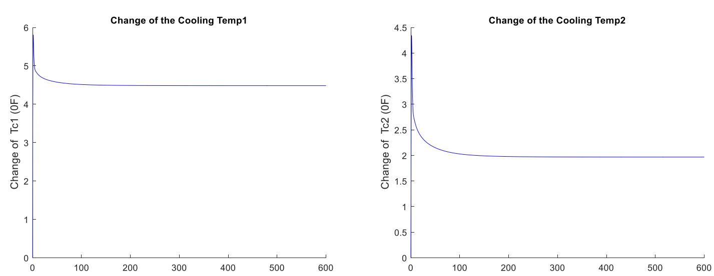
(a) (b)

Fig.10 Change of the thermal power and the fuel temperature with ∂T_{lP}(+7 ^{0}F).



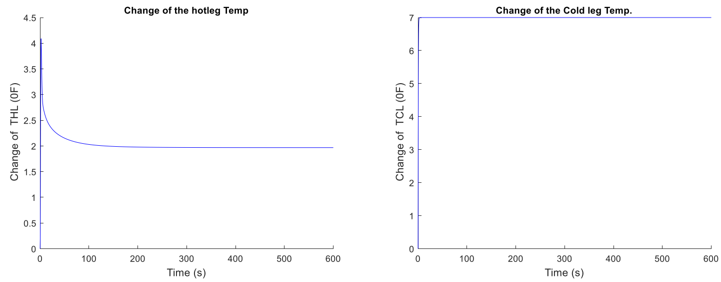
(a) (b)

Fig.11. Change of the fuel temperature and total reactivity with ∂T_{lP} (+7 ^{0}F)



(a) (b)

Fig.12. Changes of the first and second coolant temperatures with ∂T_{lP}(+7 ^{0}F)



(a) (b)

Fig.13. Changes of the hot leg and cold leg temperatures with ∂T_{lP}(+7 ^{0}F)

7. CONCLUSIONS

A new dynamic model for a SMR is developed through ODEs to represent the dynamics associated with reactor kinetics and thermal–hydraulics using MATLAB/Simulink. The developed model simulated mPower SMR for analyzing the reactor responses under specified transients. The developed model is validated by comparing the performances of the reactor core variables in response to a transient effects of increases control rod reactivity with 10 cents with reference [1]. The comparing represents the behavior of core outputs such as the fraction of power, fuel temperature, two Cooling temperatures, characteristics and hot leg and cold leg temperatures. From the comparison it is seen that the developed model is in trend as in reference [1], since the predicted relative fraction of power decreased. Additionally, the fuel and Cooling temperature increased and the hot leg temperature increased. The developed model is verified by applying many perturbations to the input parameters assumed at full power for predicating the core behavior of mPower. The verified results, show that, the developed model can be used for different transient problems and for more type of SMR reactors such as NuScale just by developing systems parameters. The model developed in this work can be utilized as a foundation for designing and testing a suitable control algorithm for reactor thermal power. Moreover, the model introduced here can be combined with relevant pressurizer and steam generator to represent for a complete analysis of the reactor core.

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