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Analysis of the control rod withdrawal speed impact on the transient behavior of Bushehr-1 NPP using Simulink software

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HIGHLIGHTS

- BNPP core neutronic and thermal-hydraulic equations in the time domain were modeled in Simulink software.
- The developed model was validated against FSAR data for the rod drop accident.
- Time domain analysis was performed using the control rod linear and non-linear models.
- The results were compared with the previous study using the transfer function approach.
- The maximum allowable control rod withdrawal speed was investigated by simulating several experiments.

ABSTRACT

In this research, the governing dynamic equations of the Bushehr NPP core are studied and modeled using Matlab (Simulink) software. The point kinetic equation with the temperature feedbacks and the fuel-coolant energy balance equations in the time domain were used for this purpose. The model is validated against the rod drop accident data available in Bushehr Nuclear Power Plant unit 1 (BNPP-1) Final Safety Analysis Report (FSAR), and they agreed. Then, this time-domain model is used to find the maximum movement speed of the control rods. For this goal, linear and non-linear rod movement equations have been modeled. In this regard, the maximum withdrawal speed of the working bank (H10) with a worth of 1.1 dollars has been investigated. Using the linear CR model, a speed limit of 9 cm.s^{-1} has been obtained to prevent the initiation of a reactor trip. The maximum speed using the non-linear model of the CR was found out to be dependent on its initial position. Thus, in three positions of the H10 bank: 100%, 80%, and 50% of the length inside the reactor, the maximum withdrawal speed values were valuated 11.5, 7.7, and 4.4 cm.s^{-1}, respectively. According to the results, among the reactor parameters including power, period, and fuel temperature, which are monitored by the reactor protection system to initiate the reactor trip, the reactor power is the limiting factor for specifying the maximum withdrawal speed. This study is performed using time domain analysis, and the obtained results are consistent with the results reported in the previous research using Laplace transform approach.

KEYWORDS
Reactor Dynamic Model
Bushehr NPP
Maximum speed of control rod
Simulink Modelling
Time domain analysis

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1 Introduction

Analyzing the dynamic behavior of nuclear reactors is very important for their safe operation (Kerlin and Upadhyaya, 2019).

In light water reactors, various factors such as delayed neutron precursor concentration, control rod worth, fuel and coolant temperature feedbacks, and fuel burnup which are themselves a function of the reactor power (or the neutron flux), would turn the core into a complex nonlinear system (Khajavi et al., 2002). Therefore, developing suitable tools for modeling the dynamic behavior of nuclear reactors is of particular importance. These tools require neutronic and thermal hydraulic modeling to predict the reactor core behavior. For the neutronic model, ideally, the time-dependent Boltzman equation can be used for modeling the kinetic core behavior. However, this is a time-consuming approach which is not practical. The large code systems use the diffusion approximation using nodal method, however, they are complicated and needs

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lots of effort for the core dynamic behavior analysis.

The point reactor kinetics equation (PKE) with temperature feedback simulates the time-dependent behavior of the nuclear reactors. It resembles a stiff system of nonlinear coupled differential equations for the power density and delayed precursor concentrations (Chen et al., 2013; Hamieh and Saidinezhad, 2012; Khoshahval and Akbari, 2020). To solve these coupled neutronic and thermal hydraulics equations, Matlab/Simulink software is an efficient tool and is employed in several studies (El_Tokhy and Mahmoud, 2017; Wang et al., 2015; El-Genk and Tournier, 2016).

To control the nuclear reactor, Control Rods (CRs) are the main elements, and they play an important role in the safety of the reactor. Several studies have been performed to investigate their impact on core behavior (Mustafa, 2021; Boromand et al., 2021; Fadaei and Setayeshi, 2009; Torabi et al., 2018). The main object of the CRs is to control the rate of neutron fissions chain inside the nuclear reactor and rate of production of steam. Thus, they adjust the level of the output electrical power. The movement speed is an important parameter in the design of Control Rod Drive Mechanism (CRDM) which is a linear stepping drive. Improper movement of CRs would lead to reactor trip and reduces the nuclear power plant (NPP) availability factor.

To avoid such unnecessary trips, or criticality accidents, the maximum movement speed of CRs should be investigated. In the previous research (Khoshahval and Ahdavi, 2016) the nonlinear dynamics equations of the unit 1 of the Bushehr Nuclear Power Plant (BNPP) core were linearized about the nominal power and the core transfer function, fuel and moderator temperature feedbacks, and the thermal-hydraulic equations were calculated in the frequency domain using Laplace transform. Moreover, the maximum CRs movement speed was studied using the developed model. However, the transfer function approach has some drawbacks. Since it is suitable for linear systems, one should adopt the linearization approximation about the operating point, which contributes to errors in simulation results. Besides, the transfer function method does not consider the initial conditions.

In this study, to avoid the limitations of the transfer function method, BNPP-1 reactor core equations are modeled in the time domain using Matlab/Simulink software. The required core data are extracted from the FSAR (FSAR, 2003) and used in the modeling. The paper outline is as follows: in section 2 the governing equations of the nuclear reactor core are elaborated. A brief description of Bushehr NPP is presented in section 3. The developed Simulink model in the time domain is described in section 4, and it is verified in section 5. In section 6, the calculation results using the linear and nonlinear CR models are presented and discussed. Finally, in section 7, the paper is concluded.

2 Material and Methods

To model the reactor core, two sets of equations including Neutronic and Thermal hydraulic models are needed. The kinetic behavior of nuclear power plants can be investigated using the continuous energy time-dependent Boltzman equation in a heterogeneous media (Bell, 1970). Neutron flux in the reactor core (comprising fuel, structure, cladding, coolant, moderator, etc.) depends on various parameters such as position, angle, energy and time. Neutrons appear at some position in the reactor as a result of a fission reaction between uranium or plutonium and a neutron from a previous generation. Moreover, neutron flux is affected by the delayed neutron precursor concentration. Thus, solving this equation over a large reactor core is not practical. Transient situations occurring in a reactor core can be analyzed through approximated solutions to the transport equation to determine neutron flux, and as a result, it is possible to reach a sufficiently precise prediction of the consequences of the transients. To simplify the neutron transport equation in terms of the angle parameter, the diffusion approximation is used. This approximation is suitable for a weak angular dependence for the neutron transport (Duderstadt and Hamilton, 1976). Also, instead of using continuous-energy calculations, the multi-group energy method is employed. Therefore, solving the diffusion equation using two to four energy groups is a common practice for core analysis codes, but is still time-consuming for transient analysis.

The P1 approximation with some extra simplification and assumptions of the neutron transport equation and Bateman equation result in point kinetic equation (PKE). It can be said that PKE is the most well-known and simplified model of reactor physics. The PKE equation assumes no shape for neutron flux and simulate it as a point. By assuming that the neutron flux spatial shape varies very little during the transient, one can calculate the temporal behavior of a nuclear reactor by using the PKEs (Kerlin and Upadhyaya, 2019; Henry, 1975). In this study, for simulating the kinetic behavior of the core (neutronic model), the PKE equation is employed. As mentioned this model has two limitations: firstly, it is assumed that the neutrons have the same speed (or energy), and secondly the spatial dependence of neutron flux is assumed to be time-independent. The PKE are as follows:

$$\begin{cases} \dfrac{\mathrm{d}n}{\mathrm{d}t} = \dfrac{\rho(t)-\beta}{\Lambda} n(t) + \displaystyle\sum_{i=1}^{6} \lambda_i C_i \\ \dfrac{\mathrm{d}C_i}{\mathrm{d}t} = \dfrac{\beta_i}{\Lambda} n(t) - \lambda_i C_i \qquad i = 1, 2, ..., 6 \end{cases} \tag{1}$$

where: C_i is the delayed neutron precursor of the i^{th} group, λ_i is the decay constant of delayed neutrons of the i^{th} group, β_i is the effective delayed neutron fraction of the i^{th} group, Λ is the mean neutron generation time, and $\rho = \dfrac{k_{\text{eff}} - 1}{k_{\text{eff}}}$ is the core reactivity.

It should be noted that core reactivity, ρ, is the input variable of the PKE. To calculate the reactor period, the following approximate formula (reciprocal period) can be used (Khoshahval and Ahdavi, 2016; Hetrick and Jarvis, 1972):

$$T = \frac{1}{\dot{\rho}} \tag{2}$$

For the safe operation of a nuclear reactor the Reactor

Protection System (RPS) is considered in its design. In order to prevent the accidents, RPS initiates the reactor trip, and to mitigate the accident consequences, it actuates the reactor Engineered Safety Features (ESF). For this goal, RPS monitors important parameters of the reactor core such as power level, period and temperature of fuel and coolant. In the Bushehr reactor, if the period value is less than 10 seconds, a trip signal will be initiated. According to Eq. (2), this condition will be approximately equivalent to increasing the slope of reactivity changes more than 0.1. It is worth mentioning that, there are also other conditions that would cause reactor scram such as exceeding the power of 107%, exceeding the applied inserted reactivity of 0.1 dollar or exceeding fuel temperature of 1883 °C. These conditions have also been evaluated in this investigation.

In order to analyze the short-term transients, in addition to the above PKE model, it is necessary to consider the following factors which influence the core reactivity:

Fuel temperature reactivity feedback: This is the change in reactivity due to the change in the temperature of nuclear fuel. This coefficient is known as the Doppler coefficient. The fuel pellet consists mainly of U-238. By increasing the temperature and changing the relative speed of neutron and nucleus, the absorption (capture) resonances of U-238 will be broadened and thus, more neutrons that are slowing down due to successive collisions, will be absorbed by U-238 nucleus. As a result, the power of the reactor will decrease due to the increase in temperature. This instantaneous negative feedback plays an important role in the compensation of the inserted positive reactivity, and acts to stabilize power reactor operations.

Coolant temperature reactivity feedback: As the moderator/coolant increases in temperature, it becomes less dense and slows down fewer neutrons, which results in a negative change of reactivity. The time for heat to be transferred to the moderator is usually measured in seconds. Therefore, comparing with the fuel temperature feedback is instantaneous, this reactivity feedback will appear within a few seconds of the core.

The temperature reactivity coefficients are defined by the following equation.

$$\alpha_T = \alpha_T^F + \alpha_T^C = \frac{1}{K}\frac{\partial K}{\partial T_F} + \frac{1}{K}\frac{\partial K}{\partial T_C} \tag{3}$$

where α_T^F is the fuel temperature reactivity coefficient, (°C^{-1}), α_T^C is the coolant temperature reactivity coefficient, (°C^{-1}), K is the effective multiplication factor, T_F is the fuel temperature, (°C), T_C is the coolant temperature, (°C). Thus, the total temperature feedback reactivity can be calculated as follows.

$$\begin{cases} \delta\rho_F = \alpha_F \delta T_F \\ \delta\rho_C = \alpha_C \delta T_C \end{cases} \tag{4}$$

$$\delta\rho_{FB} = \delta\rho_F + \delta\rho_C \tag{5}$$

where $\delta\rho_F$ is the fuel reactivity feedback value, $\delta\rho_C$ is the coolant reactivity feedback value. and δT is the temperature change value (°C).

External reactivity due to the insertion of control rods: Control rods are the main means of controlling the reactor power. These rods in BNPP-1 are made of B_4C + (Dy_2O_3-TiO_2). Insertion of a control rod causes a strong negative reactivity in the core. Therefore, to account for this reactivity, a model should be considered to convert the control rod movement into changes in reactivity. In addition, the soluble poison (boric acid) present in the coolant applies negative reactivity homogeneously to the core. In this study, the reactivity due to soluble poison is neglected since it does not play an important role in short-term transients.

The CR reactivity is usually modeled by either a linear approximation or a non-linear model. In the linear model, the reactivity value of the control rod is proportional to its insertion into the reactor core (Weaver, 1968).

$$\delta\rho_{CR} = G_r \delta x \tag{6}$$

$$G_r = \frac{\rho_w}{H} \tag{7}$$

where $\delta\rho_{CR}$ is the reactivity value of the control rod (\$), G_r is the reactivity value of the control rod per unit length (\$.cm^{-1}), δx is the insertion length (cm), ρ_w is the integral worth of the control rod (\$), and H is the effective height of the reactor core (cm). By taking the derivative from the equation, one can relate it to the movement speed.

$$\frac{\mathrm{d}\delta\rho_{CR}}{\mathrm{d}t} = G_r \frac{\mathrm{d}\delta x}{\mathrm{d}t} = G_r \nu \tag{8}$$

where ν is the movement speed (cm.s^{-1}). The position of the control rod can be obtained using the following equation:

$$x = vt + x_0 \tag{9}$$

where x_0 is the initial position of the rod.

The linear approximation is a simplified model, however, the main drawback of this model is that the reactivity worth of the control rod is considered the same throughout its length, which does not exactly correspond to reality. In fact, the reactivity worth of control rods depends on various factors including their relative position in the core, neutron flux shape, burnup and xenon concentration.

In the nonlinear model, the reactivity value of the control rod is determined using the following equation:

$$\rho(x) = \rho_w\left(\frac{x}{H} - \frac{1}{2\pi}\sin\left(\frac{2\pi x}{H}\right)\right) \tag{10}$$

where x is the rod position (cm) in the core.

This model accounts for the dependence of the rod worth to its relative position in the core. The rod worth would be small at the top and bottom of the core, and would have the highest value at the middle of the core.

For developing the thermal hydraulic model, lumped parameters models are generally considered. The simplest model in this regard would be an adiabatic model, in which the heat loss from the core is assumed to be negligible. This assumption can be suitable in a very rapid transient (Duderstadt and Hamilton, 1976). In the constant heat removal model, it is assumed that a fixed amount of the heat generated in the core is removed from it. In these

Table 1: BNPP-1 Specifications (FSAR, 2003).

Parameter	Value
Reactor nominal thermal power (MW)	3000
Coolant flow rate ($m^3.h^{-1}$)	84800
Coolant pressure at the core outlet (MPa)	15.7
Coolant temperature at the reactor inlet (°C)	291 ± 2.5
Coolant temperature at the reactor outlet (°C)	321 ± 5.0
Fuel assembly form	Hexagonal
Arrangement of fuel rods	Triangle
Number of fuel assemblies in the core	163
Number of fuel rods in the fuel assembly	311
Fuel pellet material	UO_2
Cladding material	Alloy Zr + 1% Nb
Hole diameter in the fuel pellet (mm)	1.5
Fuel pellet outside diameter (mm)	7.57
Cladding inside diameter (mm)	7.73
Cladding outside diameter (mm)	9.1
Fuel rod effective height (cm)	353
Control rod absorbing material	B_4C + ($Dy_2O_3TiO_2$)
Number of control rods	85×18

simplified models, a single effective temperature is accounted for the whole core. In the Newton's law of cooling, two different temperatures are considered for the fuel and coolant, and the heat removal from the core is assumed to be proportional to the temperature difference of fuel and coolant. In the present study, temperature effects are described by considering two distinct temperature regions: fuel and coolant (Fig. 1). This is an extended model for the Newton's law of cooling which accounts the heat transfer parameters of the core and coolant in a more detailed scheme. In this model, the energy balance between the generated heat in the fuel and the heat removed by the coolant has been considered. The heat balance equations in the fuel and coolant regions are shown in Eq. (11).

Considering that heat is generated due to fission in the fuel area and is removed by the coolant, the energy balance equation for fuel and coolant will be as follows.

$$\begin{cases} m_f C_f \dfrac{\mathrm{d}T_f}{\mathrm{d}t} = P - \dfrac{1}{R}(T_f - T_c) \\ m_c C_{pc} \dfrac{\mathrm{d}T_c}{\mathrm{d}t} = \dfrac{1}{R}(T_f - T_c) - 2WC_{pc}(T_c - T_{cin}) \end{cases} \tag{11}$$

where P is the reactor power (MW_{th}), m_f is the total mass of fuel in the core (kg), T_f is the fuel temperature (°C), C_f is the specific heat capacity of fuel ($MJ.kg^{-1}.°C$), m_c is the total mass of coolant in the core (kg), T_c is the coolant temperature (°C), C_{pc} is the coolant heat capacity at constant pressure ($MJ.kg^{-1}.°C$), W is the coolant mass flow rate passing through the core ($kg.s^{-1}$), T_{cin} is the coolant temperature entering the core (°C), and R is the thermal resistance between fuel and coolant, ($°C.MW^{-1}$).

3 Bushehr-1 NPP Specification

Bushehr NPP (unit 1) is a WWER1000/V446 type with four coolant loops. It consists of 163 hexagonal fuel assemblies each containing 311 fuel rods and 18 guiding channels for control rods or burnable poisons. It employs ten control banks. In nominal power, all groups except group H10 are in the top position above the reactor. The main specifications of BNPP-1 are listed in Table 1. Neutronic and thermal hydraulic parameters are also extracted from the FSAR document (FSAR, 2003) and presented in Table 2.

Table 2: BNPP-1 core parameters (FSAR, 2003).

Parameter	Value
β_{eff}	0.0074
β_1	2.563E-04
β_2	1.524E-03
β_3	1.401E-03
β_4	3.084E-03
β_5	1.110E-03
β_6	2.650E-04
Λ (s)	3.20E-05
α_F (°C)	-2.11E-05
M_f (kg)	79.84
C_f ($MJ.kg^{-1}.°C$)	3.71E-04
R ($°C.MW^{-1}$)	0.1217
P_0 (MW)	3000
T_{f0} (°C)	671.31
Position of H10 (%)	80
λ (s^{-1})	0.0286
λ_1 (s^{-1})	1.27E-02
λ_2 (s^{-1})	3.17E-02
λ_3 (s^{-1})	1.55E-02
λ_4 (s^{-1})	3.11E-02
λ_5 (s^{-1})	1.4
λ_6 (s^{-1})	3.87
H (cm)	353
α_c ($°C^{-1}$)	-13E-5
M_c (kg)	10,422
C_{cp} ($MJ.kg^{-1}.°C$)	5.85E-03
W ($kg.s^{-1}$)	16,704
T_{in} (°C)	291
T_{c0} (°C)	306.34
ρw ($)	1.11

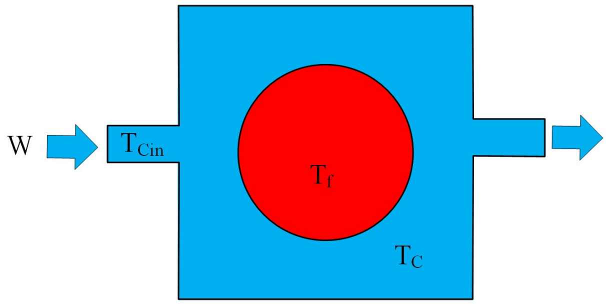

Figure 1: Thermal hydraulic core model.

4 Simulink Modelling

Neutronic and thermal-hydraulic constants of the equations 1 to 10 related to the Bushehr-1 reactor were extracted from reference (FSAR, 2003) and used in the Simulink model. The overall arrangement of the blocks including CR movement, reactivity converter, core and the temperature feedback models are depicted in Fig. 2.

As mentioned above, the core model, which employs the PKE in the time domain is shown in Fig. 3. In this study, the 6-group precursor model is used. The fuel and coolant temperature feedback equations (Eq. (11)) are modeled in Simulink and shown in Fig. 4. As can be seen in this figure, the input of this block is the reactor power value (MW) and the output is the sum of fuel and coolant temperature reactivity feedbacks. For the conversion of the temperature to reactivity, the reactivity temperature coefficients: α_f and α_c (Eq. (4)) are used. For compliance to the core model, the reactivity conversion block is used to convert the unit of the inserted reactivity from dollars to $\Delta K/K$.

The linear model of the control rod movement is shown in Fig. 5. Equations 6 to 9 are employed for converting the rod movement to reactivity changes. The rod movement length is modeled using the saturation dynamic block. Finally, the non-linear CR model (Eq. 10) is depicted in Fig. (6).

5 Verification of the model

To verify the Simulink model of the BNPP-1 reactor core, the rod drop accident (FSAR, 2003) has been simulated. In this accident, a control rod of 62 pcm worth falls into the reactor core within 4 seconds, and the results obtained are compared with the BNPP-1 Final Safety Analysis Report (FSAR) (FSAR, 2003) in Figs. 7 to 9. Under the influence of the CR drop, the reactor power suffered a sudden drop after the fall. It reaches the minimum value just 4 seconds after the drop. The power dip is calculated to be 0.93 of the initial power (nominal power) and is reported as 0.89 in FSAR. After this sudden drop, the reactor power increases during the next 6 seconds, and according to the simulation, it will reach the value of 0.94. This number is reported as 0.91 in FSAR. Therefore, the simulation result for calculating the reactor power value is about 3.3% higher than the values reported in FSAR. This difference is due to the fact that the PKE is actually an approximation of the diffusion equation that ignores spatial dependencies. It should be noted in the previous study using Laplace transform, the dip and final values for the reactor power were calculated to be about 0.94 and 0.942 respectively. Thus, the time domain analysis to obtain reactor power has led to closer results to FSAR data compared to the previous research. This could be due to the error caused by the linearization of the equations at the nominal power, which was done to obtain the transfer function using the Laplace transform.

Figure 8 depicts the reactivity variations in the reactor core during the rod drop accident. This curve can be divided into two parts. In the first part, the core reactivity is rapidly decreased by the dropping of the rod (applying the external reactivity). This negative reactivity is induced on a short time interval and reaches a dip value. After this point, in the second part, the negative reactivity induced by the rod drop is compensated by the positive reactivity caused by the fuel and coolant temperature feedbacks. This reactivity compensation takes place gradually over a longer time interval.

Figure 9 shows the fuel temperature variation during this transient. This is an important core safety parameter which should be specified to avoid fuel damage during transients. Comparing with reactivity (Fig. 8) and power (Fig. 7) curves, the fuel temperature takes a longer time to decrease from its initial value of 675 °C. The final fuel temperature value in FSAR is 642 while it is calculated to be 648 °C by the model having less than 1% error. As can be seen, the simulation results agree with the reported data in FSAR.

6 Results and Discussion

The withdrawal of the control rods causes positive reactivity to the reactor core. As mentioned earlier, this positive reactivity affects important reactor parameters such as power, period, and fuel temperature which are monitored for initiation of reactor trip command by the Reactor Protection System (RPS). In this section, the maximum withdrawal speed of CRs will be analyzed using the developed model. For this goal, first, the relevant trip parameters and their set values should be considered. BNPP-1 RPS will trip the reactor in the following conditions (FSAR, 2003):

- Reactor period becomes less than 10 seconds.
- The power value reaches 107% of the nominal powe.
- The applied reactivity value exceeds 0.1 dollars.
- Fuel temperature exceeds 1883 °C.

Therefore, to determine the maximum withdrawal speed of the control rods, the speed threshold that leads to the reactor trip should be calculated in linear and non-linear models.

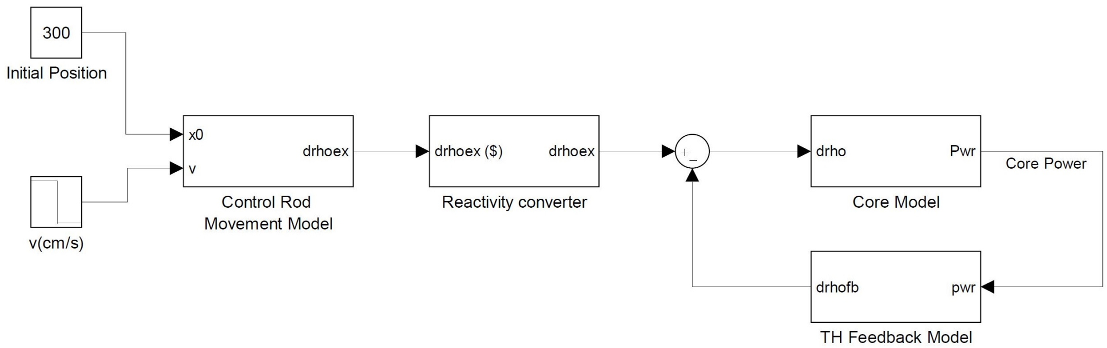

Figure 2: Neutronic and thermal-hydraulic model of Bushehr reactor (unit 1) core in Simulink.

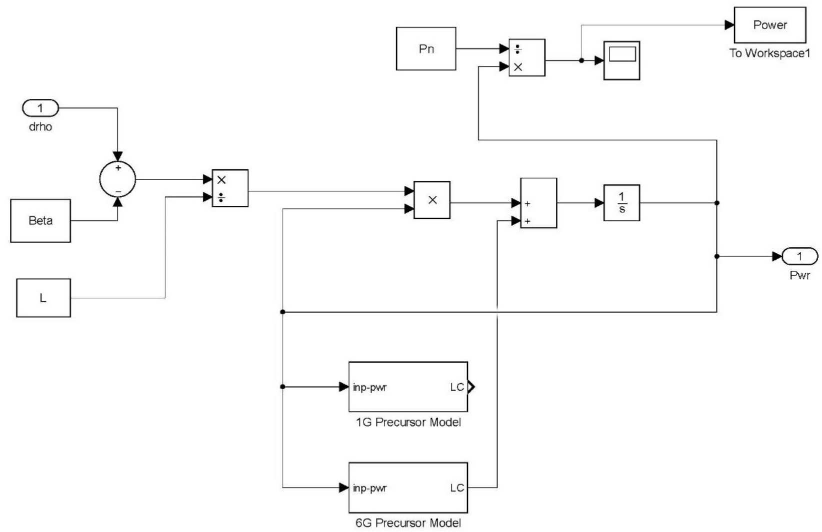

Figure 3: Point kinetic equation modeling in time domain in Simulink.

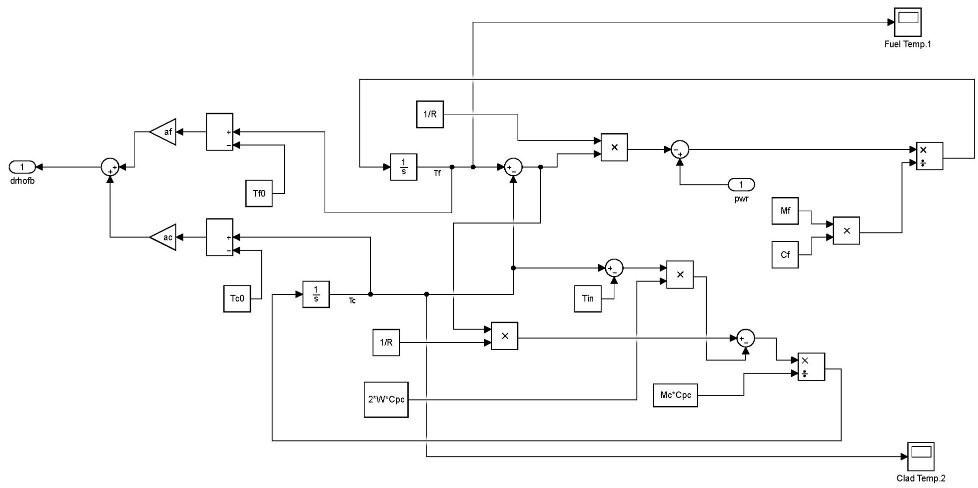

Figure 4: Fuel and coolant temperature feedback model.

6.1 Linear Model

In order to investigate the rod withdrawal speed limit, an experiment is simulated using the CR linear model. In this experiment, the initial position of the rod is 282.4 cm (80%) inside the core and it is pulled out by 30 cm. It should be noted that at $x = 0$, the rod is completely outside and for $x = H$, the rod is entirely inside the core. The reactivity worth of the rod is equal to 1.11 dollars (equal to the worth of CRs of group H10). It is notable that in the linear model, the worth of the CR is assumed to be the same throughout its length, thus, the initial position

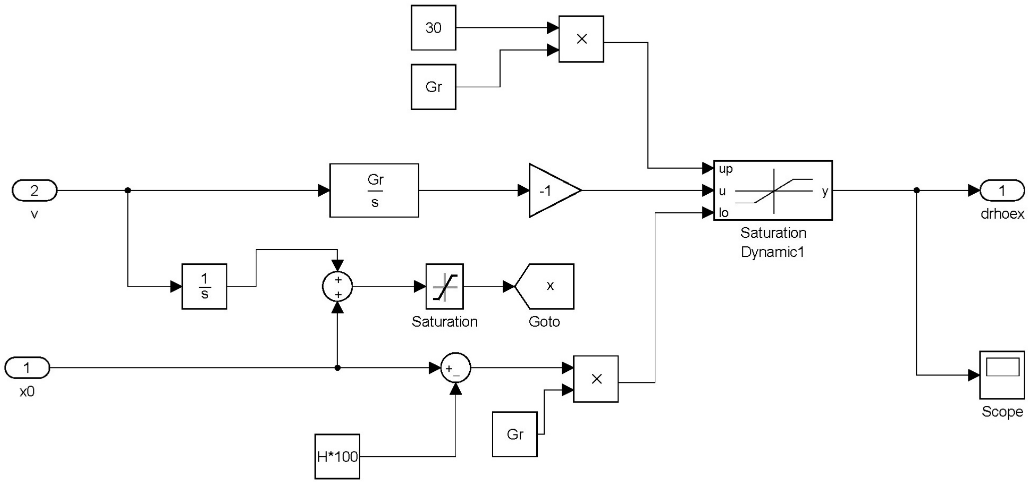

Figure 5: The linear model of the control rod in Simulink.

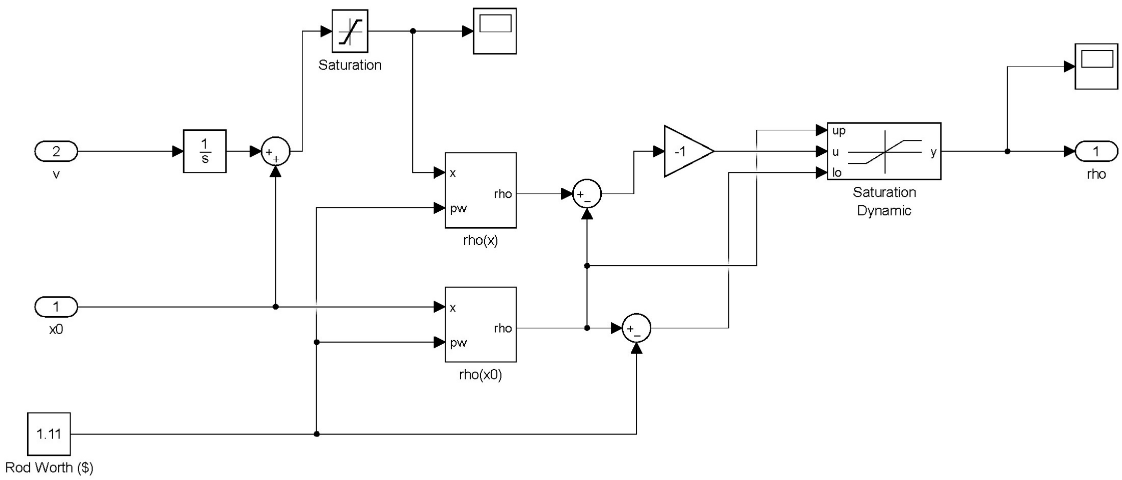

Figure 6: The non-linear model of control rod in Simulink.

of the rod will not affect the result. The variation of core relative power versus time in this simulation is depicted in Figure 10. As can be seen in this figure, the reactor power will increase immediately after pulling out the rod and will reach a peak value. The time and the height of this peak depend on the speed of CR withdrawal. By moving the control rod at a speed between 3 to 10, the time to reach the peak power will be between 3 to 10 seconds and the peak value will vary between 5.7 to 7% of the relative power. For speeds more than 9 cm.s^{-1}, the peak will exceed the nominal power by 7%, and thus the trip signal will be initiated.

The core reactivity, reciprocal period, and fuel temperature variations versus time are also shown in Figs. 11 to 13. The condition of mentioned parameters during this transient is always in the safe range, and the limiting factor in this experiment is the rated power value. Thus, the withdrawal speed limit is 9 cm.s^{-1}. It is worth noting that, in the previous research (Khoshahval and Ahdavi, 2016), which was done using Laplace transform approach, this value was calculated to be 8 cm.s^{-1}.

6.2 Nonlinear Model

The nonlinear behavior of the reactivity value due to the CR insertion into the reactor core is modeled using Eq. (10). In order to have a better understanding of this behavior, the integral and differential CR worth is shown in Fig. 14 using Eq. (10). According to Fig. 14, the highest value of reactivity worth of the CR takes place when the rod is located in the middle of the core. In this case, pulling out the rod will lead to applying the highest positive reactivity to the core. Therefore, the maximum withdrawal speed of CR without tripping the reactor will reach the lowest value in this case.

To investigate the maximum allowable rod speed in this situation, an experiment is simulated using the developed model which employs the nonlinear CR equation. In this experiment, the CRs of group H10 with a worth of 1.1 dollars is considered to be in the middle of the core at x_0 = 176.5 cm (50% in the core), and it is pulled up by 15.3 cm. The variation of reactor power in terms of time during this transient is shown in Fig. (15). As can be seen in this figure, after pulling out the control rod, the reac-

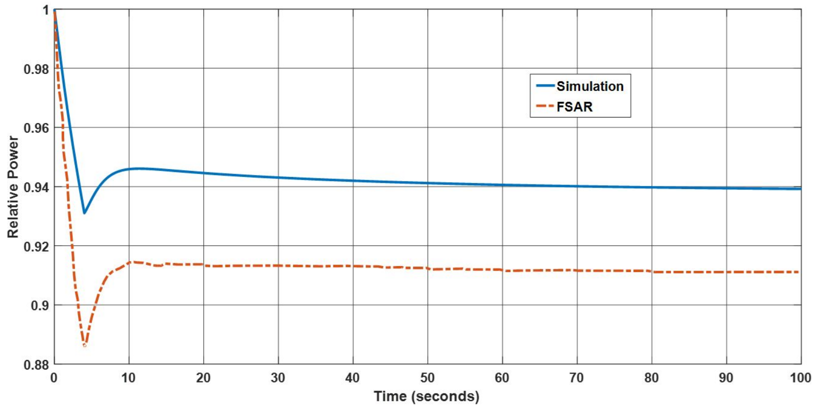

Figure 7: Relative power variation during rod drop accident.

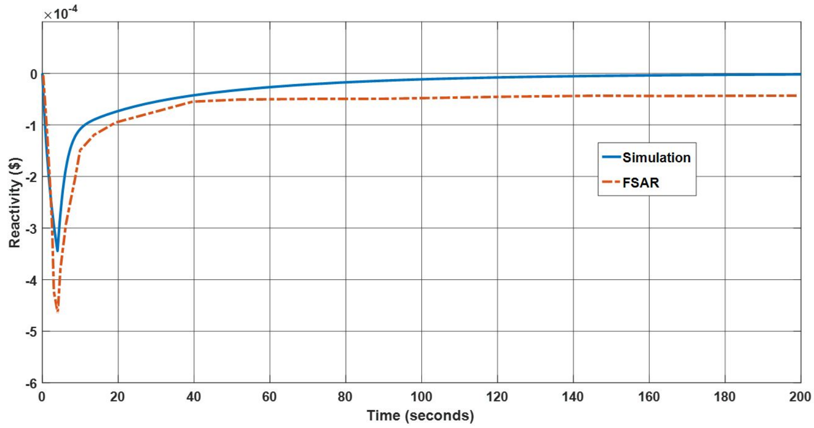

Figure 8: Core reactivity variation during rod drop accident.

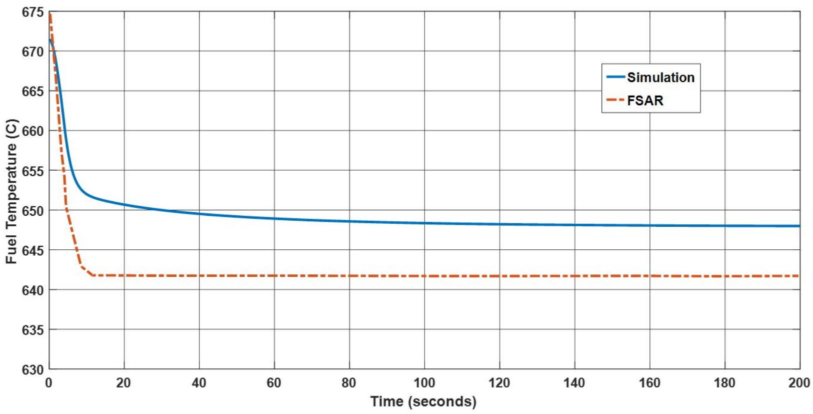

Figure 9: Fuel temperature variation during rod drop accident.

tor power increases rapidly and reaches a peak value. As mentioned, the amplitude and time of this peak depend on the rod withdrawal speed. Compared to the previous experiment (pulling out 30 cm of the rod which is positioned at 80% inside the core), the peaks occur faster (less than 5 seconds). This is due to the fact that 15 cm of the rod at the 50% position has a higher worth. The simulation has been done for the rod speeds between 3 cm.s^{-1} to 8 cm.s^{-1}, and the peak height has changed from 6.5 to 8 percent. For speeds higher than 4.4 cm.s^{-1}, a trip signal will be initiated by RPS. The changes in reactivity, period, and fuel temperature during this period are shown in Figures 16 to 18. According to these figures, the values of these parameters during this transient are all below the trip set point, and the limiting factor for determining the rod withdrawal speed is the reactor power.

As mentioned before, the CR withdrawal speed depends on its initial position in the core. In order to have

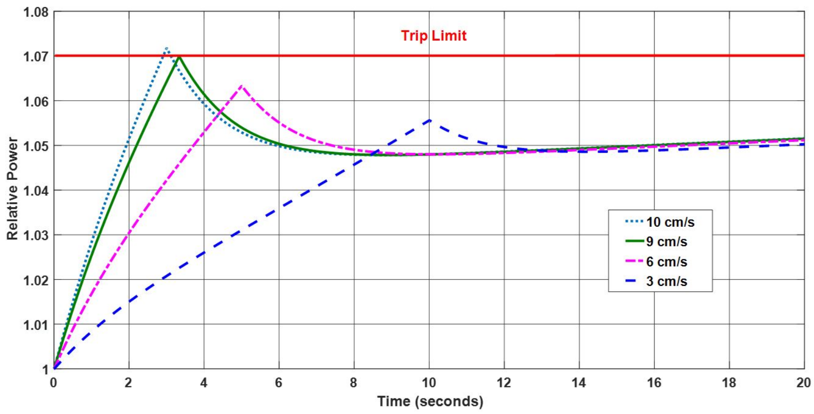

Figure 10: Relative power changes versus time using CR linear model.

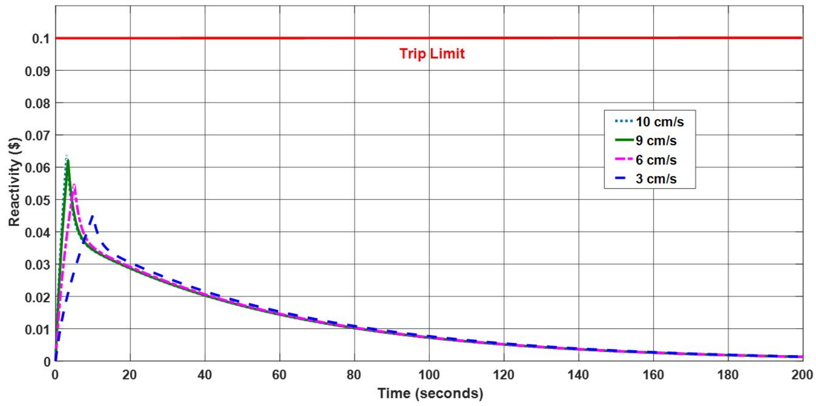

Figure 11: Core reactivity variation versus time using CR linear model.

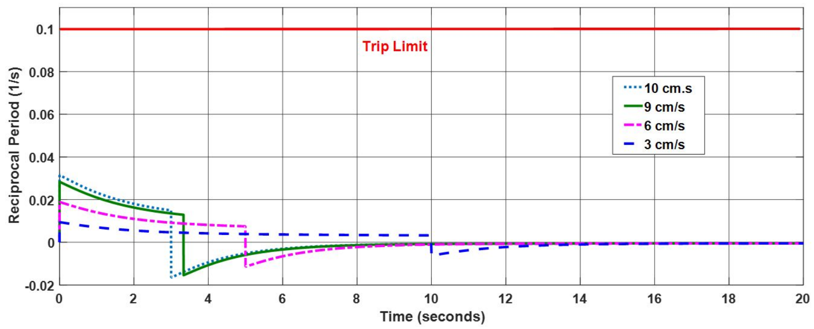

Figure 12: Reciprocal period variations versus time using CR linear model.

a better understanding of this issue, two extra tests have been simulated. In the first test, the rod is positioned 100% in the core, and 86.5 cm is pulled out. In the next test, the rod is positioned 80% inside the core and 31.5 cm is pulled out. The maximum speed values have been calculated for these two tests and are shown in Table 3.

Table 3: Maximum rod speed for different initial positions.

Initial Position (cm)	176.5 cm	282.4 cm	353 cm
(% in core)	(50%)	(80%)	(100%)
Withdrawal Length (cm)	15.3	31.5	86.5
Maximum Speed (cm.s^{-1})	4.4	7.7	11.5

According to the results listed in this table, due to the lower worth values of the CR in the positions of 80 and 100%, higher withdrawal speed limits (7.7 and 11.5 cm.s^{-1}, respectively) have been obtained.

7 Conclusions

In this research, the equations governing the dynamic behavior of BNPP-1 reactor core were studied. These equations were modeled in the time domain using Simulink software. The developed model was validated against the available data in BNPP-1 FSAR for the rod drop exper-

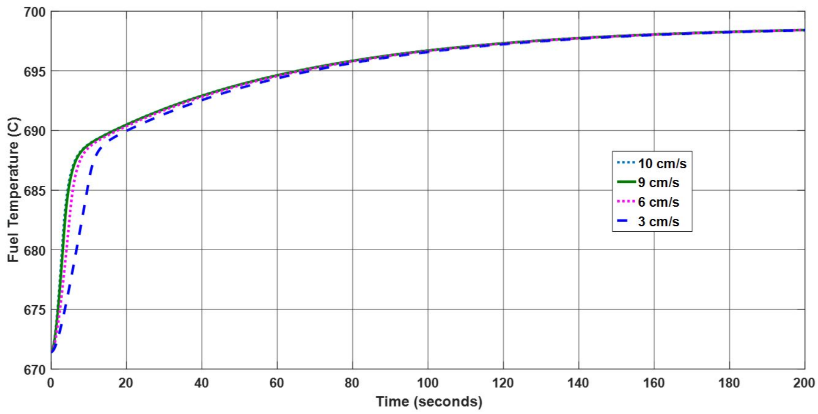

Figure 13: Fuel temperature variation versus time CR linear model.

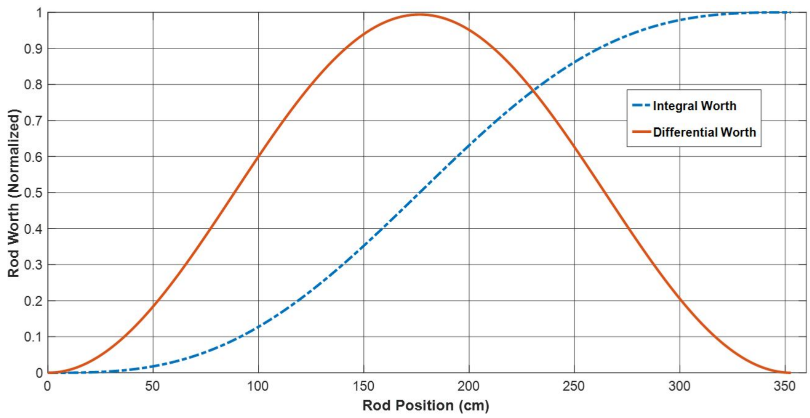

Figure 14: Integral and Differential CR Worth.

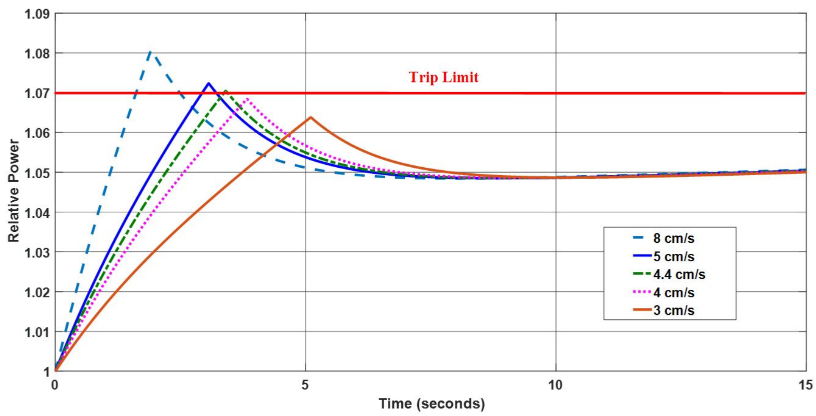

Figure 15: Relative power variations versus time using a CR nonlinear model.

iment. The simulation results were in good agreement with the FSAR data. Moreover, in comparison to the results reported in the previous research using the transfer function method by Laplace transform, the time-domain analysis for calculating the reactivity in the core has led to closer results to the manufacturer's data.

The developed model has been used to study the maximum CR movement speed without a reactor trip. For this purpose, two linear and non-linear models were considered to simulate the movement of the control rods. Accordingly, two experiments were simulated to determine the maximum permissible speed of the CRs of group H10 with a total worth of 1.11$. Based on the results using the linear model, the maximum speed was obtained to be 9 cm.s^{-1}. This value was calculated to be 8 cm.s^{-1} in the previous research using the transfer function approach.

In the second experiment, the nonlinear model was used to determine the maximum speed when it is placed in the middle of the core (50%). In this position, because the worth of the rod is the highest value, the allowable

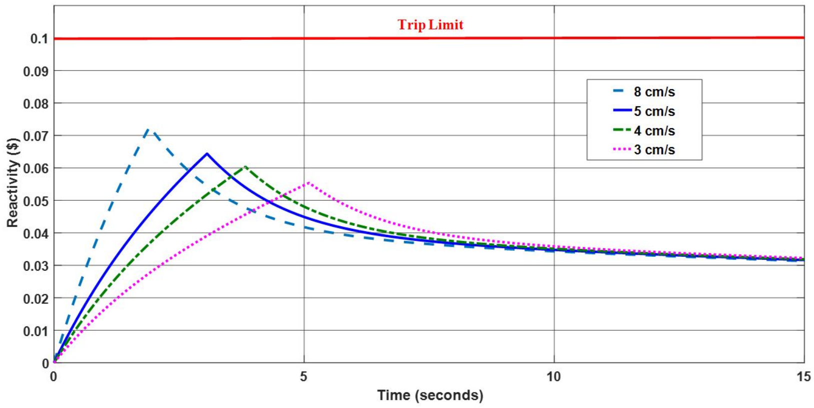

Figure 16: Core reactivity variation versus time using CR nonlinear model.

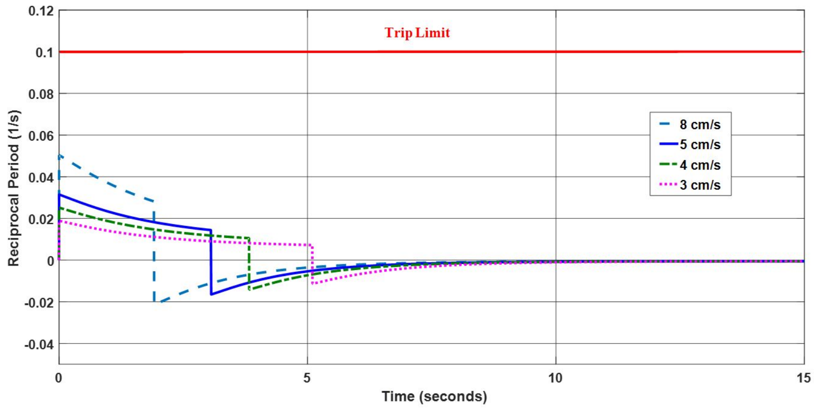

Figure 17: Reciprocal period variations versus time using CR nonlinear model.

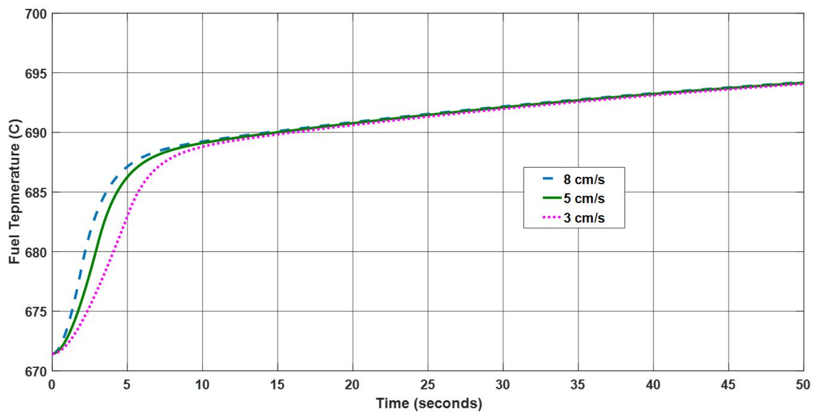

Figure 18: Fuel temperature variation versus time CR nonlinear model.

speed was reduced to 4.4 cm.s^{-1}.

Finally, due to the dependence of the CR's maximum speed on its initial position in the core, two extra tests were simulated at positions 80% and 100% inside the core. The maximum withdrawal speeds for these tests were calculated to be 7.7 cm.s^{-1} and 11.5 cm.s^{-1}, respectively. The results indicate that a margin of about 120% is considered for choosing the speed of control rods to avoid unintended trip by the BNPP-1 manufacturer.

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

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