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Simulation of Pressurized Water Reactor under Specified Reactivity Accidents

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***Abstract*—In Nuclear power plants, Reactivity-Induced Accidents category is very important since, they part of the licensing design basis accidents analyses regulatory required for pressurized water reactors. Rod Ejection Accidents are part of Reactivity-Induced Accidents since they are induced due to the failures of its operating mechanism with the power evolution driven by reactivity insertion. In this research, a new dynamic mathematical model is developed to simulate a of H. B. Robinson Pressurized Water Nuclear Reactor for predicting its dynamic response under Rod Ejection Accidents specified transients, that are suggested by reference [1]. The developed model comprises a point kinetics description of neutronics and thermodynamics in the reactor core, pressurizer, plenums, hot and cold legs and utilizes a recirculation (U-tube) steam generator that composed the nuclear steam supply system model. The developed model is validated by comparing with previous literature of reference [2], the results nearly agreement. The results illustrate that the developed model represents the dynamic features of real-physical systems and are capable of predicting responses due to small and large perturbations of external control rod reactivity. The developed model can be used for other many different transient problems and also can be simulated other type of PWR reactors by developing input parameters data.**

Keywords—Nuclear Steam Supply System; Reactivity-Induced Accidents; pressurized water reactors; A Rod Ejection Accident.

I. INTRODUCTION

Reactivity is a fundamental quantity that expresses the deviation of the reactor criticality, so it expresses the imbalance between the number of neutrons that are absorbed and the number of neutrons that appear due to nuclear fission. For safety and the integrity of core in conditions of accidents, the monitoring of the reactivity may be accomplished using simulation with computer codes or the instrumentation and control system data [3]. Analysis of reactivity accidents is an important part of the safety study in any nuclear reactor since, effect on the total energy released, which can produce power peak, increase fuel, clad and coolant temperatures [3].

Reactivity insertion events in power reactors can be divided broadly into: Control system failures, Control element ejections, Events caused by coolant/moderator temperature and void effects and Events caused by dilution or removal of coolant/moderator poison [4].

The rod ejection and its associated addition of reactivity to the core occur within about (0.1 sec) in the worst possible scenario; the actual time depends on reactor coolant pressure and the severity of the mechanical failure. Simulations and understandings of such accidents are key objectives for safety and regulatory authorities around the world.

A specific safety criterion relative to rod ejection was defined in the 1970s based on American tests, limiting the deposited energy (enthalpy) in the fuel (cal/g) during the reactivity transient. In the early 1990s, however, following the Chernobyl accident and, more especially, in view of the gradual increase in fuel assembly burnups, the international scientific community began to question the validity of this criterion, for moderate burn ups [5]. The U.S. NRC established a Phenomena Identification and Ranking Tables (PIRT) panel for defining and evaluating the phenomena happening during rod ejection accidents for pressurized water reactors (PWRs) and boiling water reactors (BWRs) that containing high-burnup fuel in 2001. Main goal of that document is presentation the most relevant parameters of fuel and reactor core, during accident conditions, that are important for plant safety analysis. In [6] selected TRACE/PARCS codes as a computational tool which is development of the U.S. NRC research program Code Application and Maintenance Program (CAMP). In reference [7] PIRT calculated all phenomena during REA, the most relevant parameters were investigated in that paper, using computational model used a VVER-1000/V-320 type of reactor with hexagonal lattice geometry. Moreover, Japan and France for CABRI reactor, developed research programs, including IPSN's tests that aimed to improve understanding of the physical phenomena that could lead to cladding leaks, and the ejection of fuel fragments into the reactor coolant system, which could be harmful to core cooling. The OECD report cited in the reference [8] is a State-Of-the-Art Report on the knowledge of RIAs as of 2010 [9].

Many researches developed models to study REA such as in reference [10] where, the REA analysis is performed at both critical Hot zero power (HZP) and Hot full power (HFP) conditions. Moreover, the impact of fuel-cladding gap-heat transfer model during the REA is also analyzed. The concluded from the performed simulations have shown that there is a high margin against cladding melting, fuel and the maximum fuel enthalpy is found to be far from fuel-cladding failure threshold and there is a significant impact of the models used to describe the fuel-cladding gap heat transfer on the prediction of key safety parameters such as the maximum fuel enthalpy. Finally, in reference [11] perform a sensitivity analysis of a REA that called CRW event in a temperature gas-cooled reactors (mHTGRs) model using code RELAP5-3D with point kinetics feedback to demonstrate the impact of uncertainty in heat transfer and reactor kinetic parameters. That study addresses a need highlighted by the Nuclear Regulatory Commission (NRC) for transient fuel testing by quantifying the impact of uncertainty in heat transfer and reactor kinetic parameters and by generating potential boundary conditions for transient testing of conventional mHTGR fuel.

In this research, the responses neutronics and thermo hydraulics are analyzed during inserting reactivity in the pressurized water reactors through using a new developed modeling. This model is developed for the analysis of the temperatures and pressures typical of H. B. Robinson nuclear power reactor, during reactivity insertion transient REA. The reactivity transients' analyses have been made considering the insertion of many different input Steps. All steps are suggested as in the reference [1] for the analysis of accidents due to reactivity insertion. After that many specified reactivity transients are simulated to examine reactor dynamics with the developed model due to many steps of reactivity increase due to others REA accidents. In Section 2; represents the methodology of the developed model that is applied in the H. B. Robinson Nuclear power reactor, and the validation of the developed is represented in section 3. The results of the developed model applied under specified transients are described and dissection in section 4 and conclusion is presented in Sections 5.

II. THE METHODOLOGY

A PWR steam supply system (NSSS) model is developed in this research that contains a reactor core, primary coolant system, pressurizer and steam generator as depicted in [12]. The required data are taken from design and final safety analysis reports for The H. B. Robinson Nuclear Power Plant [2]. In the developed model, a reactor core, and primary coolant system to typical analysis of a thermal fluids system involves conservation of mass, momentum and energy. The balance accounts for the heat transfer from the nuclear fuel to the coolant using Mann's model [12] that utilizes two coolant lumps for every fuel lump as seen in fig. 1.

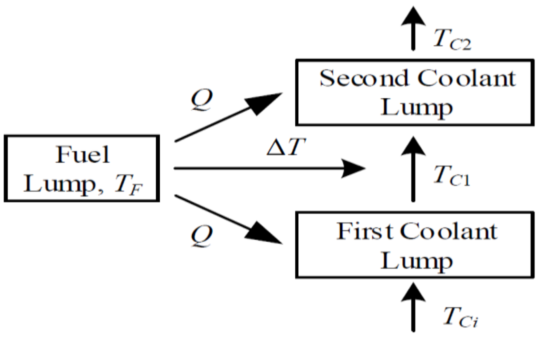


Fig. 1. Mann's heat transfer model

A. *Reactor core Neutronics*

The point kinetics equations for reactor power, one delayed neutron precursor and the core reactivity are represented as flowing:

$$\frac{d\,\partial P_{th}}{dt} = \frac{\partial \rho_t - \beta}{\Lambda}\partial P_{th} + \lambda\,\partial C \quad (1)$$

$$\frac{d\,\partial C}{dt} = \frac{\beta}{\Lambda}\partial P_{th} - \lambda\,\partial C \quad (2)$$

$$\frac{d\,\partial \rho_t}{dt} = \alpha_F\,\partial T_F + \frac{\alpha_C}{2\Lambda}\partial T_{C1} + \frac{\alpha_C}{2\Lambda}\partial T_{C2} + \partial \rho_{ext} \quad (3)$$

Where:

P_{th} Reactor thermal power
C Delayed neutron precursor concentration
β Delayed neutron fraction
λ Decay constant
ρ_t Total reactivity
α_F Reactivity feedback from fuel
α_{Ci} Reactivity feedback from cooling nodes
ρ_{ext} Control rod reactivity
T_F Average temperatures of the fuel lumps
T_{C1} Temperature of the first coolant lump
T_{C2} Temperature of the second coolant lump

B. *Reactor Core Thermodynamics*

Typical analysis of a thermal fluids system involves conservation of mass, momentum and energy. This analysis is simplified by first assuming that the reactor coolant is at constant density, pressure and in mass flow rate. The energy balance accounts for the heat transfer from the nuclear fuel to the coolant two coolant lumps for every fuel lump as shown in Fig. 1. The linearized to calculate the parameters values changes with time of form of the reactor thermal hydraulics is given by equation from (4) to (6) are being used with their parameters value are provided by [11] as shown in table 1.

$$\frac{d\,\partial T_F}{dt} = \left(\frac{fP_0}{m_f c_{pf}}\right)\partial P_{th} + \frac{U_{fc}A_{fc}}{m_f c_{pf}}(\partial T_{C1} - \partial T_F) \quad (4)$$

$$\frac{d\,\partial T_{C1}}{dt} = \frac{U_{fc}A_{fc1}P_0}{2m_{c1}c_{pc}}(\partial T_f - \partial T_{C1}) + \frac{2W_p}{m_{c1}}(\partial T_{Lp} - \partial T_{C1}) \quad (5)$$

$$\frac{d\,\partial T_{C2}}{dt} = \frac{U_{fc}A_{fc2}P_0}{2m_{c2}c_{pc}}(\partial T_f - \partial T_{C1}) + \frac{2W_p}{m_{c1}}(\partial T_{C1} - \partial T_{C2}) \quad (6)$$

Where:

c_{pc} Coolant heat capacity
c_{pf} Fuel heat capacity
m_c Mass of coolant in core
m_{c1} Coolant mass node 1
m_{c2} Coolant mass node 2
f Fraction of the total power directly deposited in the fuel
U_{fc} Heat transfer coefficient from fuel to coolant
A_{fc} Effective heat transfer surface area
m_f Mass of fuel,
Wp Primary coolant mass flow rate

TABLE 1. PARAMETERS FOR DETERMINING MODEL NODAL

Core Variable	***value***	***unit***
P_o	2200	MW_{th}
F	0.974	–
B	0.0064	-
Λ	$1.6x10^{-5}$	S
cpc	1.39	Btu/lb.°F
α_c	$-2.0x10^{-4}$	(Δk/k)/°F
αp	$-1.3x10^{-5}$	(Δk/k)/Ps
m_c	$1.3x10^{-5}$	Lb
m_c	406050	Lb
cpf	0.059	Btu/hr.ft.°F
Afc	42460	ft^2
U_{fc}	179	Btu/hr.ft.°F
m_f	222739	Lb
Λ	$1.79X\ 10^{-5}$	lb/hr
Wp	$101.5x10^8$	lb/hr

C. *Pressurizer*

The pressurizer regulates the primary coolant pressure, which is controlled by spraying water from the cold leg and actuating electric heaters, and serves to dampen fluctuations. A two-region pressurizer model [11]. The governing equations for the pressurizer are presented as follows:

$$\frac{d\,\partial P_p}{dt} = B_1\,\partial P_p + B_2\,\partial W_w + B_3 q \quad (7)$$

Where:

P_p The pressure in the pressurizer
W_w Mass of flow of water into and output of the pressurizer
q The rate of heat addition to the pressurizer fluid with electrical heater

The values of B_1, B_2, and B_3 for the H. R. Robinson Nuclear Plant are: B_1, = $-1.913\ X\ 10^{-6}$ (sec^{-1}), B_2 = $7.02\ 1\ X\ 10^{-3}$ (psi/lb) and B_3 = $2.1726\ X\ 10^{-4}$ [psi/(kW sec)] [2].

D. *Plenums, Hot and Cold Legs*

In the developed model, there are two piping sections (the hotleg & coldleg) and four plenums (upper & lower and inlet & outlet steam generator) in the model. In reactor plenums, complete mixing is assumed during normal transients [12]. The energy conservation equations are applied on the four plenums as following:

$$\frac{d\,\partial T_{LP}}{dt} = \frac{W_p(\partial T_{CL} - \partial T_{LP})}{m_{LP}} \quad (8)$$

$$\frac{d\,\partial T_{UP}}{dt} = \frac{W_p(\partial T_{C2} - \partial T_{UP})}{m_{UP}} \quad (9)$$

$$\frac{d\,\partial T_{CL}}{dt} = \frac{W_p(\partial T_{OP} - \partial T_{CL})}{m_{CL}} \quad (10)$$

$$\frac{d\,\partial T_{HL}}{dt} = \frac{W_p(\partial T_{UP} - \partial T_{HL})}{m_{HL}} \quad (11)$$

$$\frac{d\,\partial T_{IP}}{dt} = \frac{W_p(\partial T_{HL} - \partial T_{IP})}{m_{IP}} \quad (12)$$

Where:

Wp Primary coolant mass flow rate
m_{LP} Mass of water in the lower plenum
m_{uP} Mass of water in the upper plenum
m_{cL} Mass of water in the coldleg
mH_L Mass of water in the hotleg
m_{IP} Mass of water in primary of SG
T_{IP} Inlet SG primary water temperature
T_{LP} Reactor lower plenum temperature
T_{CL} Cold-leg temperature
T_{up} Reactor upper plenum temperature
T_{OP} Temperature of primary coolant in the steam generator outlet plenum
T_{HL} Hot-leg temperature,
T_{IP} Temperature of primary coolant in the steam generator inlet plenum.

E. Steam Generator

SG Model is developed consists of the three lumps as shown in [14]:

1. Primary fluid lump (PRL).
2. Heat conducting tube metal lump (MTL).
3. Secondary fluid lump (SFL).

The Governing Equations for SG Model:

$$\frac{d\,\partial T_p}{dt} = \frac{W_p}{m_{pI}}(\partial T_{IP} - \partial T_p) - \frac{U_{pm}A_{pm}}{m_{pI}c_{pI}}(\partial T_p - \partial T_m) \quad (13)$$

$$\frac{d\,\partial T_m}{dt} = \frac{U_{pm}A_{pm}}{m_m c_m}(\partial T_p - \partial T_m) - \frac{U_{ms}A_{ms}}{m_m c_m}(\partial T_m - \partial T_s) \quad (14)$$

$$\frac{d\,\partial P_s}{dt} = D_1\,\partial T_m + D_2\,\partial P_s \quad (15)$$

$$\partial T_s = \frac{\partial T_{sat}}{\partial P_s}\,\partial P_s \quad (16)$$

Where:

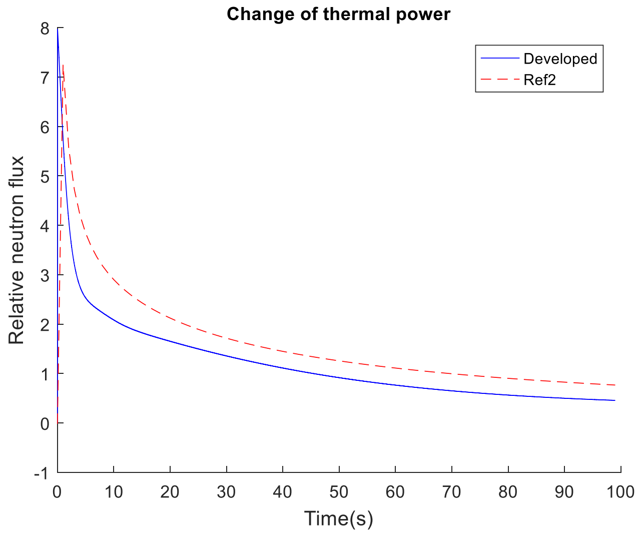
T_m Average temperature of tube metal lump.
C_{pl} Specific heat of primary water in the steam generator
T_p Temperature of primary coolant node in the steam generator
m_{pI} Mass of primary water in the steam generator
C_{pm} Specific heat of tube metal
U_{ms} Effective heat transfer coefficient between the tube metal and secondary fluid lumps
A_{ms} Heat transfer area between the tube metal and secondary fluid lumps,
U_{pm} Effective heat transfer coefficient between primary water and tube metal lumps
A_{pm} Heat transfer area between the primary water and tube metal lumps
m_m Mass of tube metal
T_s Bulk mean temperature in the secondary lumps
P_s Steam pressure

The values of D_1, D_2, for the H. R. Robinson Nuclear Plant are: 1.349 & -0.2034 respectively and $\frac{\partial T_{sat}}{\partial P_s}$ is the slope of the straight-line approximation of the curve of Ts at a gain step [2].

III. VALIDATION OF THE DEVELOPED MODEL

A comparison of reactor core dynamic simulations between the developed model results and previously published results in reference [2] are considered. Considering the case of positive reactivity insertion simulated by adding a 0.000032 (Δk/k) as a step reactivity increasing at t = 0 sec, the comparison between change of the thermal powers and the fuel temperatures are shown in fig. 2a and fig. 2b.

The validation results show that in both models the transient change in power and fuel temperatures of the NSSS is in the same trend.



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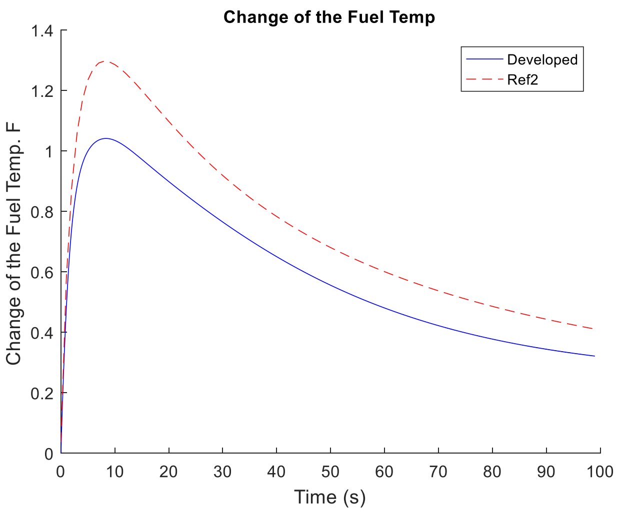


(b)

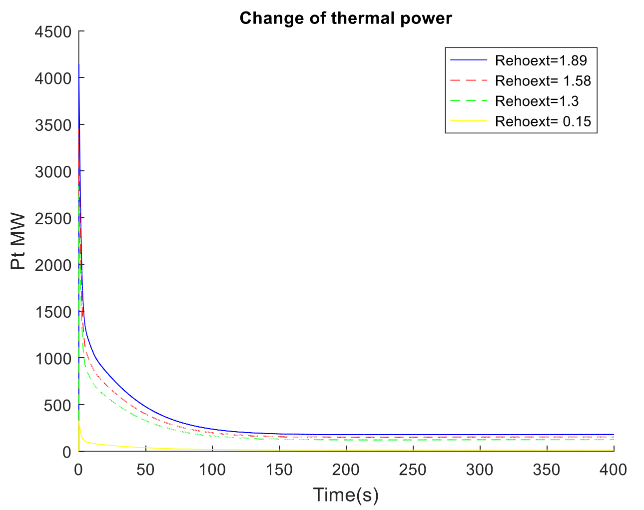
Fig.2. Change in reactor thermal power & the fuel temperature

IV. Reactor Response for Specified Reactivity REA Accidents

The developed model calculated distributions of energy and temperatures of the core and all systems are predicted during many postulated control rod ejection accidents. Core initial conditions Reactivity insertion (Δρ) Peak enthalpy increase that investigator reference [1] with four values [1.89$, 1.30$, 1.58$, 0.15$]. The study is done for H. B. Robinson Nuclear Plant a typical three-loop PWR, in which the core consisted fuel assemblies of 15×15 design. The ejection of a fully inserted control rod was postulated at the end of a reactor operating cycle, while the core was held at hot zero power conditions.

The analysis of pulse width during rod ejection accidents are carried out that can be used to help in designing experiments to test fuel behaviour under reactivity-initiated accident conditions.

The unexpected reactivity increase can be due to a control rod withdrawal or a sudden pump start up. Such events had been simulated by adding the values [1.89, 1.30, 1.58, 0.15] as a step reactivity increasing at t = 0 sec as reference (IAEA). The analysis used calculations based on the developed model of a pressurized water reactor.



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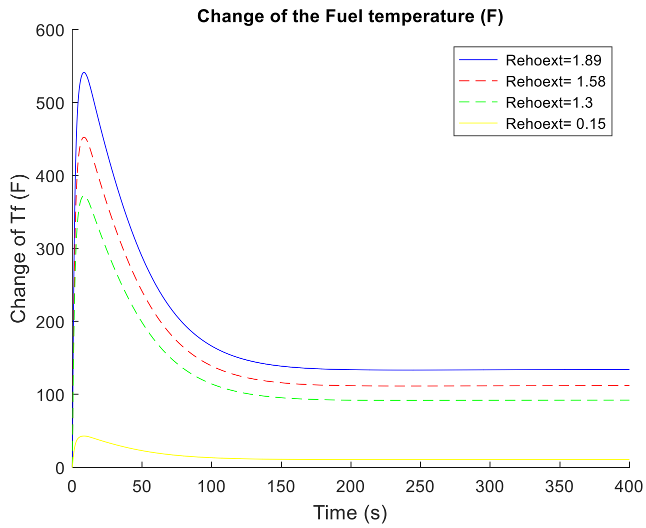

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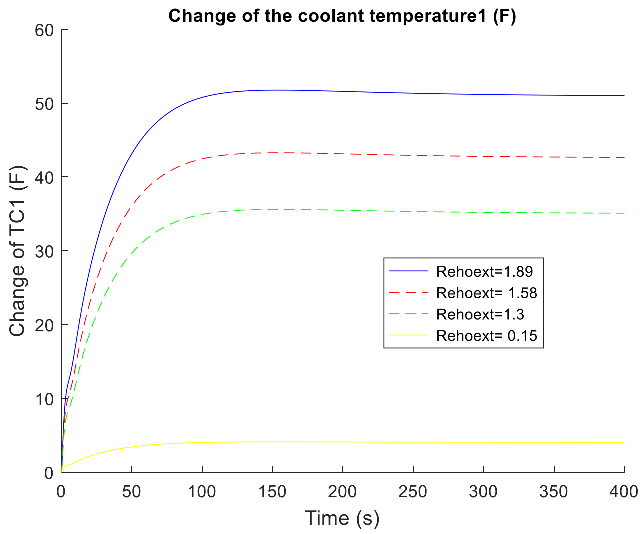
Fig. 3 Change in reactor thermal power & the fuel temperature



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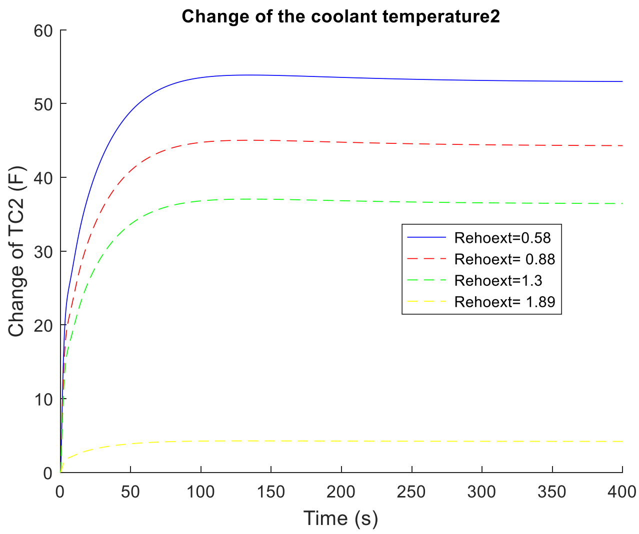

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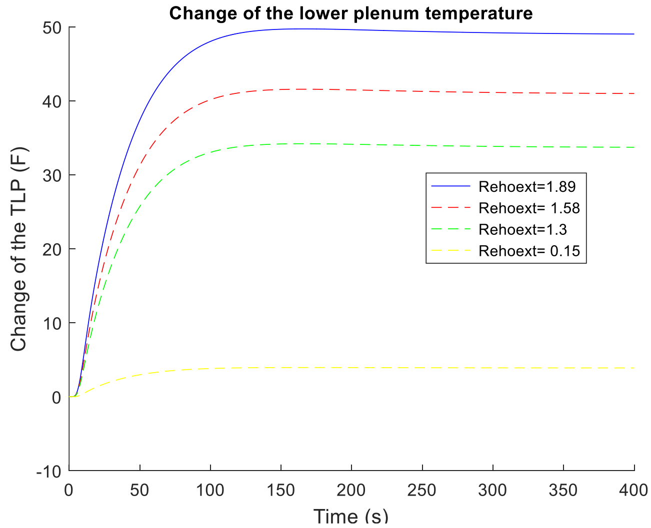
Fig. 4 The first coolant temperature & second coolant temperature



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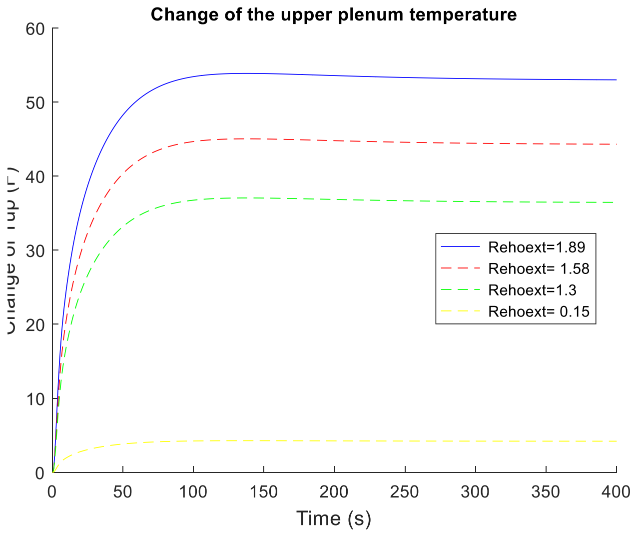

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Fig. 5 The change of lower & upper plenum temperatures

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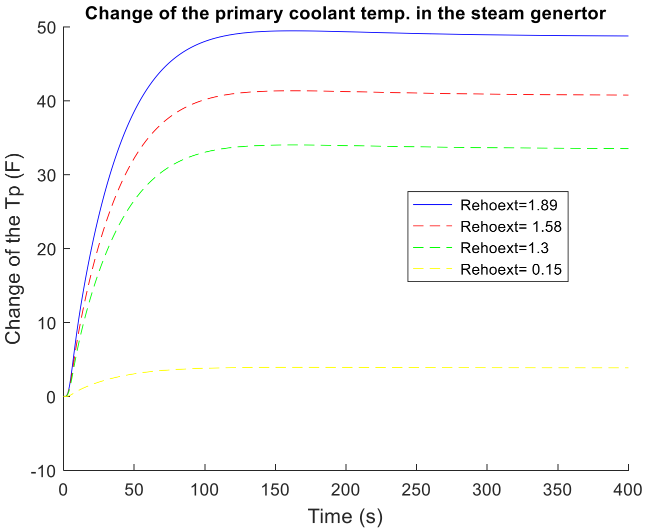

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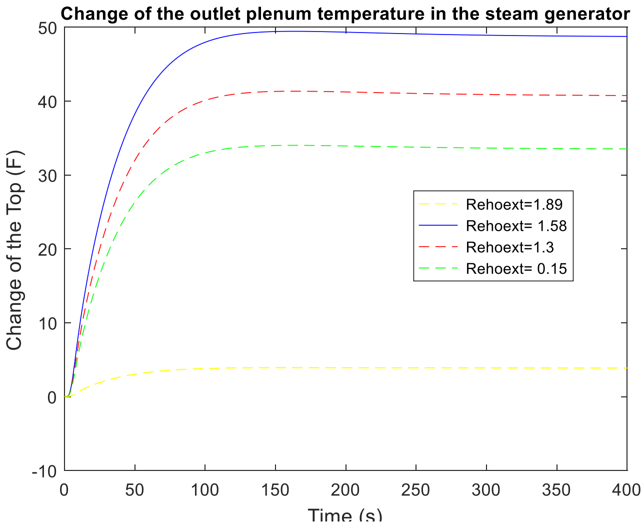

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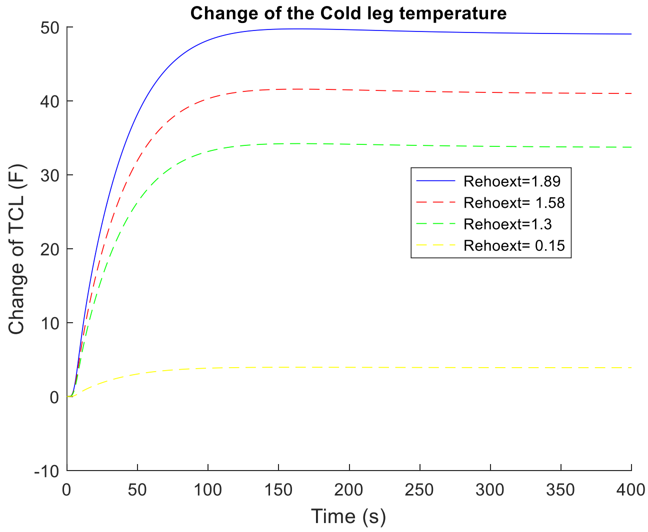
Fig. 6 The change of primary coolant & Metal temperature in SG



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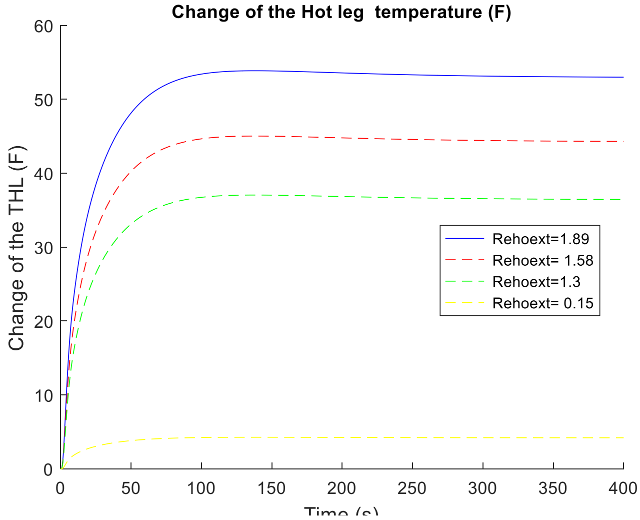

(b)

Fig. 7 The change of clod-leg & Hot-leg temp.

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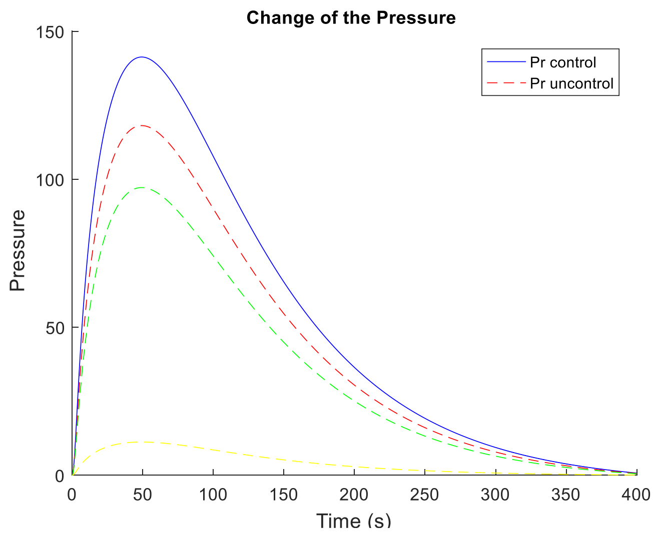

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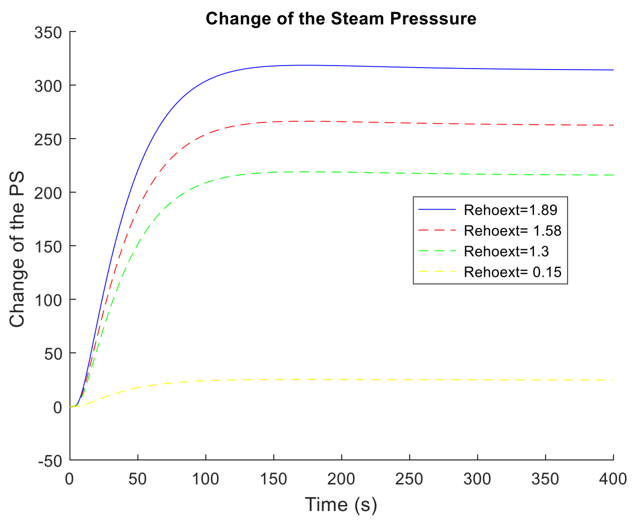

(b)

Fig. 8 The change of Pressure in the primary side & steam pressure

The Results from the figs. (3-8) show that, the increase in the reactivities act cause an increase in the fission rate and neutron flux, congruently, an initial prompt jump in the fractional reactor thermal power, as shown in fig. 3a. Subsequent the increased power generation, the fuel temperature increases and more heat is transferred from the fuel region to the primary coolant in the core. As the fuel temperatures increases not quickly, the Doppler effect provides a negative reactivity and decreases the power change. As the coolant temperature increases due to fuel heat-up, the moderator temperature change starts making additional changes in reactivity feedback. The reactivity feedbacks bring a steady-state stable power level (P_{th}) of different values that increased with increase the reactivity inserted. Thus, the PWR is stable under a reactivity insertion without any control action but has sudden increase that in danger that can be reach to sever accidents. The new steady-state fuel (T_F), and also show that pulse width varied inversely with the maximum increase in local fuel enthalpy and this is consistent with simple analytical models as shown in fig. 3b but the two Coolant temperatures T_{C1} &T_{C2} temperatures rise by , respectively, as shown in Fig. 4a and 4b. but it varies proportionally with the maximum increase in local coolant, the steam generator enthalpy for plenums and reach to steady states after about 100 second as shown in figs. (3-8). As shown in fig. 8a, the pressure in the pressurizer increases and then decreases to low values after 400 seconds but the steam pressure as shown in fig. 8b increases then reaches to steady state also about 100 seconds.

V. CONCLUSION

Reactivity Initiated Accident Reactivity (RIA) initiated accident covers an unwanted increase in fission rate and reactor power. The furthermost severe design basis Reactivity-Initiated Accident (RIA) considered for a pressurized water reactor in terms of uncontrolled nuclear reaction called the control Rod Ejection Accident (REA). The reactor power increase could damage the reactor core, and could lead to disruption of the normal performance of the reactor. In Pressurized Water Reactors (PWR), control Rod Ejection Accidents (REAs) can occur due to mechanical failure of the control rod drive mechanism or its housing, such that the reactor coolant system pressure would cause the ejection of a partially or fully inserted control rod, and drive the shaft to its fully withdrawn position. If the reactor is operating at or close to the critical position, the consequences of this mechanical failure include rapid reactivity insertion and core power increase, together with an asymmetric core power distribution. This may lead to localized fuel rod damage. The fuel temperature rapidly increases resulting in fuel pellet thermal expansion, and in very severe cases, failure in the cladding. For this reason, resistibility to REA accident is an important parameter for nuclear reactor safety and licensing.

In this research, a new mathematical dynamic modeling of Nuclear Steam Supply System (NSSS) in a Pressurized Water-type Nuclear Reactor PWR is developed for predicting the dynamic response under specified reactivity for REA. The NSSS model comprises a point kinetics description of neutronics and thermodynamics in the reactor core, pressurizer, plenums, hot and cold legs and utilizes a recirculation (U-tube) steam generator. In addition, a steady-state control program for the reactor is developed. Adequacy the complete NSSS PWR model are transient tested and validated for many perturbations of reactivity due to control rod position the validation is realized by comparing the results with other model, the results are nearly in agreement with the plant parameters available in previous literature [2].

The developed Model predicts the outputs performance in response time during many specified transients. The specified simulated transient are effects of increases reactivity with many REAs due to reactivities values Core initial conditions, Reactivity insertion ($\Delta\rho$) Peak enthalpy increase that investigator in reference [1] with four values [1.89$, 1.30$, 1.58$, 0.15$]. From the results of REA, From the results, it is found that the responses for the change of the power can give sudden increase that can lead to sever accident since, it is sudden reached to high value that can lead to core melt. The developed model can be used for other many different transient problems and also can be simulated other type of PWR reactors by developing input parameters data.

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