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Research Article

Design and Development of Neutronics and Thermal Hydraulics Modeling Code for ACP1000 Nuclear Reactor Dynamics in Lab VIEW

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Abstract: An advanced neutronics and thermal hydraulics nuclear code, called GNTHACP code, is designed and developed in LabVIEW as Graphical Neutronics and Thermal Hydraulics toolkit for 1100 MWe Advanced Chinese PWR (ACP-1000) nuclear power plant. The reactor neutronics model is developed using a nonlinear point reactor kinetics model, while the reactor thermal hydraulics model is developed based on nonlinear fuel and coolant temperature dynamics. The heart of the GNTHACP code is the control rod reactivity model. Control rod reactivity banks are comprised of four power compensation banks G1, G2, N1, N2 and one temperature compensation bank R. The reactivity control configuration of these banks is highly nonlinear, complex and challenging in nature. The control rod reactivity model as a function of G1, G2, N1, N2 and R banks is optimized using two distinct techniques. The control rod reactivity model is optimized using Simplex Linear Programming (SLP) technique under constraints of reactor power as safety limit and control rod speed as maximum speed limit in LabVIEW. The control rod reactivity model is also optimized and investigated using nonlinear Sequential Quadratic Programming (SQP) technique under same constraints in LabVIEW. All the models are integrated and the state-of-the-art virtual instruments (VIs) are designed for cost function optimization, configuring models and calibration of model parameters in LabVIEW. The integrated model as graphical coupled neutronics and thermal hydraulics modeling code is optimized and validated against the Final Safety Analysis Report (FSAR) and different parameters of interest are investigated and proved within design limits as reported with CORCA and CORTH benchmark nuclear codes. The proposed code is stable, highly efficient and accurate as compared to other nuclear codes.

Keywords: Reactor Neutronics, Thermal Hydraulics, Linear Optimization, Nonlinear Optimization, ACP1000, Nuclear Power Plant.

1. INTRODUCTION

Nuclear reactor codes are designed, developed and used for industry-standard modeling of nuclear reactor cores for transient, safety and accident analyses. A reactor kinetics and dynamics model was developed in detail for the PWR type nuclear reactor by Johnson *et al.* [1], while the coupled transient neutronics calculations were performed for molten fast reactor by Laureau *et al.* [2]. An educational simulator for PWR neutronics was developed by Lam [3]. The educational tool was further extended for PWR neutronics with special emphasis on transient and safety studies by Mollah *et al.* [4]. The reactor neutronics simulator is developed in LabVIEW by Hakim *et al.* [5]. The research is extended to thermal hydraulics studies of PWR by Ibrahim [6]. LabVIEW based Graphical User Interface (GUI) is developed for thermal hydraulics Reactor Excursion and Leak Analysis Program (RELAP) code for PWR by Macedo *et al.* [7]. A 3D neutron diffusion code is developed by Park *et al.* [8] for PWR neutron kinetics studies. A reactor dynamics code is developed for PWR studies for three different ratings of PWRs including ACP1000 using deep learning technique by Malik *et*

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al. [9]. Coupled neutronics and thermal hydraulics analysis is performed for nuclear reactor by Rais *et al.* [10]. Research is further extended for neutronics and thermal hydraulics sub-channel analysis of PWR by Ribeiro *et al.* [11] using Computational Fluid Dynamics (CFD) techniques. Point reactor kinetics model is optimized using Particle Swam Optimization (PSO) technique by Mousakazemi [12]. PWR control rod drive mechanism is addressed with emphasis on 3D modeling and analysis by Tanaka [13]. Parameters of PWR steam cycle are optimized by simplex optimization technique by Wang *et al.* [14]. Sequential quadratic programming is used for large scale nonlinear problems by Boggs *et al.* [15]. Neutronics analysis is performed for research reactor with emphasis on safety parameters by Torabi *et al.* [16].

In the present research work, a third generation PWR nuclear reactor ACP1000 is addressed for design, modeling, analysis and simulation purposes. The techniques addressed above [13-16] are adopted for this research work. Accordingly, a novel state-of-the-art coupled neutronics and thermal hydraulics graphical code is developed with novel control rod reactivity optimal models using simplex linear programming and nonlinear sequential quadratic programming algorithms in LabVIEW for the first time for ACP1000 nuclear power plant. These new models based on SLP and SQP optimization techniques are more stable, efficient, faster and accurate as compared to other numerical techniques and Industry Standard Toolset (IST) nuclear codes such as CORCA and CORTH codes. SQP exhibits excellent stability and convergence for solving large-scale optimization problems. SQP can find high Net Present Value (NPV) after about few iterations. CORCA and CORTH codes are coupled through traditional master slave coupling. CORCA and CORTH codes do not incorporate the comprehensive modeling of control rods for advanced ACP1000 reactor dynamics. Average coolant temperature dynamics predicted by CORTH code is overestimated. Therefore, GNTHACP Code is a one step ahead towards nuclear code development as well as numerical optimization based toolkit development for ACP1000 reactor dynamics in LabVIEW. Various parameters used in the present research work are described as following:

n_r = Relative Reactor Power
ρ = Net Reactivity
$\rho_{INTERNAL}$ = Internal Reactivity
$\rho_{EXTERNAL}$ = External Reactivity
ρ_{FUEL} = Reactivity due to Fuel
$\rho_{MODERATOR}$ = Reactivity due to Moderator
ρ_{CR} = Reactivity due to Control Rods
β = Delayed Neutron Fraction
Λ = Average Neutron Life Time
λ = Decay Constant
C = Precursor Concentration
$G_{CR-BANK}$ = Worth of Control Rod Bank
M_F = Mass of Fuel
C_F = Specific Heat Capacity of Fuel
M_C = Mass of Coolant
C_{PC} = Specific Heat Capacity of Coolant at Constant Pressure
W = Flow Rate of Coolant
R = Thermal Resistance
T_F = Temperature of Fuel
T_C = Temperature of Coolant
T_{IN} = Temperature of Inlet

2. MATERIALS AND METHODS

2.1 ACP1000 Neutronics Modeling in CORCA Code

The CORCA Code is a two-group two-dimension fine-mesh static neutron diffusion and core burn-up calculation code. CORCA Code could be used in light water moderated PWRs. The code is capable to do calculations in both partial (1/8, 1/4, 1/2) and whole geometry. Baffle, thermal shield and reflector can be described in detail by the code. Replacement of assemblies' location is allowed in the code, which is frequently used in refueling calculations.

All the nuclear design parameters calculated, such as Neutron Effective Multiplication Factor (NEMF called K_{eff}), Moderator Temperature Coefficient (MTC), Total Peaking Factor (Fq) and Control rod worth, etc., are evaluated, and the accuracy of all the parameters is found the same as that of international comparable nuclear design codes, meeting the requirements of engineering design. The fuel temperature coefficient is calculated by performing two-group X-Y calculations using the CORCA Code. Moderator temperature is held constant and the power level is varied. Spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of reactor power.

2.2 ACP1000 Thermal Hydraulics Modeling in CORTH Code

The objective of reactor core thermal design is to determine the maximum heat removal capability in all flow sub-channels and show that the core safety limits, as presented in the technical specifications are not exceeded while compounding engineering and nuclear effects. The thermal design considers local variations in dimensions, power generation, flow redistribution and mixing. The design is made using the CORTH computer code which is a three dimensional core sub-channels of variable size and form connected together. It determines in a very general way the steady state and transient flows of a fluid flowing in separate or connected channels. It is, thus, a suitable tool for the thermal-hydraulic analysis of reactor cores or experimental loops with heating rod bundles (limits of heat extraction from the core and in particular as it affects the critical heat flux). The CORTH Code gives all thermal-hydraulic variables in every node of the mesh: temperature of coolant, pressure, enthalpy, quality, void fraction, heat flux and flow-rate. It determines the margin with regard to the critical heat flux phenomenon.

2.3 Neutronics Modeling of ACP1000 Reactor Core

The neutronics modeling of ACP1000 reactor core is carried out using point reactor kinetics model with six precursor groups. The coupled relative neutron power and precursor concentrations are given as [1]:

$$\frac{dn_r(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda} n_r(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \qquad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n_r(t) - \lambda_i C_i(t) \qquad (2)$$

where the symbols having their usual meanings.

Six group precursors are chosen because these are representative groups in terms of half-lives of fission fragments and provides the sufficiently accurate neutron dynamics.

The net reactivity is given as:

$$\rho(t) = \rho_{INTERNAL}(t) + \rho_{EXTERNAL}(t) \qquad (3)$$

Internal reactivity is given as:

$$\rho_{INTERNAL}(t) = \rho_{FUEL}(t) + \rho_{MODERATOR}(t) \qquad (4)$$

where the symbols having their usual meanings.

Now, assuming the reactor is controlled with control rod banks. So, the external reactivity is given as:

$$\rho_{EXTERNAL}(t) = \rho_{CONTROLROD}(t) = \rho_{CR}(t) \qquad (5)$$

where the symbols having their usual meanings.

2.4 Reactor Thermal Hydraulics Modeling of ACP1000 Reactor Core

The fuel temperature dynamics is given as [6]:

$$M_F C_F \frac{dT_F(t)}{dt} = n_r(t) - \frac{1}{R}[T_F(t) - T_C(t)] \qquad (6)$$

The average reactor coolant temperature dynamics is given as:

$$M_C C_{PC} \frac{dT_C(t)}{dt} = \frac{1}{R}[T_F(t) - T_C(t)] - 2WC_{PC}[T_C(t) - T_{IN}(t)] \qquad (7)$$

where the symbols having their usual meanings.

2.5 Control Rod Reactivity Modeling of ACP1000 Reactor Core

The control rod reactivity is given as [13]:

$$\frac{d\rho_{CR}(t)}{dt} = G_{CR-BANK} \frac{dx_{CR}(t)}{dt} = G_{CR-BANK} v_{CR}(t) \qquad (8)$$

where $x_{CR}(t)$ and $v_{CR}(t)$ are the control rod bank position and control bank velocity respectively.

Amongst all models adopted in this research work, the control rod reactivity model is the most complex because it is comprised of four power compensation banks G1, G2, N1, N2 and one temperature compensation bank R which are configured in a highly nonlinear fashion.

2.6 Optimization of Control Rod Reactivity Model

The control rod reactivity model is optimized by the following two techniques:

i) Simplex linear optimization technique
ii) SQP nonlinear optimization technique

2.6.1 *Simplex Linear Optimization Problem Formulation*

Simplex linear optimization algorithm is used to optimize the control rod reactivity problem described in equation (8). This technique is the best choice with linear constraints which is the core advantage of this method [14].

$$\min \quad f(v_{CR})$$

Subject to the linear constraints:

$$g(v_{CR}^j) \geq 0, \quad j = 1,2,3,4$$
$$v_{CR}^1 \leq v_{CR}^2$$
$$v_{CR}^2 \leq v_{CR}^3$$
$$v_{CR}^3 \leq v_{CR}^4$$
$$v_{CR}^j \leq v_{CR}^{j,\max}$$
$$n_r \leq n_r^{SafetyLimit}$$

where v_{CR} is a j-dimensional variable vector, $f(v_{CR})$ is the objection function or cost function, $v_{CR}^{j,\max}$ is the upper bound of variable v_{CR}^j and $g(v_{CR}^j)$ is the jth inequality constraint of control rod reactivity optimization problem.

2.6.2. *SQP Nonlinear Optimization Problem Formulation*

SQP nonlinear optimization algorithm is used to optimize the control rod reactivity problem described in equation (8). SQP exhibits excellent stability and convergence for solving large-scale optimization problem of control rod reactivity comprising of several banks operating in complex configuration. This technique is the best with nonlinear constraints.

$$\min \quad f(v_{CR}(t))$$

Subject to the nonlinear constraints:

$$g(v_{CR}^j(t)) \geq 0, \quad j = 1,2,3,4$$
$$v_{CR}^1(t) \leq v_{CR}^2(t)$$
$$v_{CR}^2(t) \leq v_{CR}^3(t)$$
$$v_{CR}^3(t) \leq v_{CR}^4(t)$$
$$v_{CR}^j(t) \leq v_{CR}^{j,\max}$$
$$n_r(t) \leq n_r^{SafetyLimit}$$

where symbols having their usual meanings in nonlinear domain.

Now, the nonlinear control rod reactivity optimization problem is converted into QP sub-problem by means of gradient and nonlinear constraints. The solution space is obtained by series of QP sub-problems as [15]:

$$g(v_{CR}(t)) = \nabla f(v_{CR}(t))$$
$$\min \quad \frac{1}{2} m^{kT} H^k m^k + g^T (v_{CR}(t) m^k$$

where H^k the Hessain matrix of Lagrangian is function and m^k is the solution of SQ sub-problem.

Subject to the nonlinear constraints:

$$v_{CR}^1(t) + \nabla v_{CR}^1(t) m^k \leq v_{CR}^2(t) + \nabla v_{CR}^2(t) m^k$$
$$v_{CR}^2(t) + \nabla v_{CR}^2(t) m^k \leq v_{CR}^3(t) + \nabla v_{CR}^3(t) m^k$$
$$v_{CR}^3(t) + \nabla v_{CR}^3(t) m^k \leq v_{CR}^4(t) + \nabla v_{CR}^4(t) m^k$$
$$v_{CR}^j(t) + \nabla v_{CR}^j(t) m^k \leq v_{CR}^{j,\max} + + \nabla v_{CR}^{j,maz}(t) m^k$$
$$n_r(t) + + \nabla n_r(t) m^k \leq n_r^{SafetyLimit} + \nabla n_r^{SafetyLimit}(t) m^k$$

2.7 Framework of Neutronics and Thermal Hydraulics Modeling Code of ACP1000 Nuclear Reactor

The overall framework of neutronics and thermal hydraulics model consists of point reactor kinetics model, thermal hydraulics model of fuel and coolant and control reactivity model with both linear and nonlinear optimization techniques. The framework of ACP1000 neutronics and thermal hydraulics modeling is shown in Figure 1. All the models are coupled dynamic in nature solved and computed in parallel computing framework. LabVIEW is selected as programming platform because it is the best choice for nuclear code as stand-alone product development with powerful excellent user friendly GUI. However, initially, the coupled integrated model is first analyzed with simplex linear programming (SLP) and then with sequential quadratic programming (SQP).

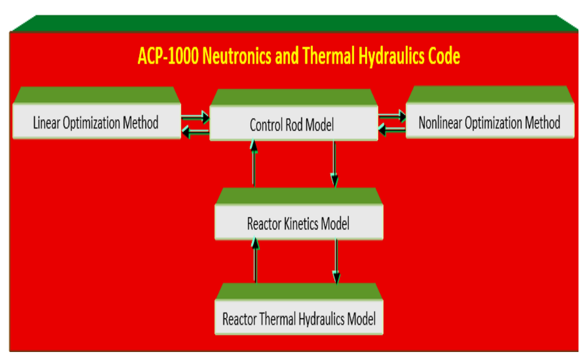

Fig. 1. Framework of ACP1000 neutronics and thermal hydraulics modeling.

3. RESULTS AND DISCUSSION

In the present investigations, the entire modeling and linear as well as non-linear optimization work is carried out in LabVIEW. Results are discussed in the following sections:

3.1 Modeling and Linear Optimization in LabVIEW

In the present research work, a virtual instrument (VI) is designed for linear simplex optimization as shown in Figure 2. The design of linear model optimization VI for control rod reactivity is shown in Figure 3. The linear constrained optimization problem is solved and various parameters of interest are analyzed. The initial value of control rod position is x_o = 357.1875 cm. The control rod speeds are returned with four set of speeds as shown in Figure 4.

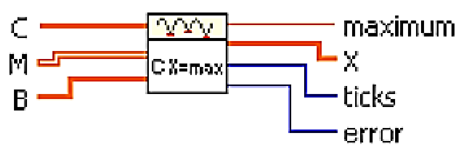

Fig. 2. Design of linear simplex optimization VI in LabVIEW

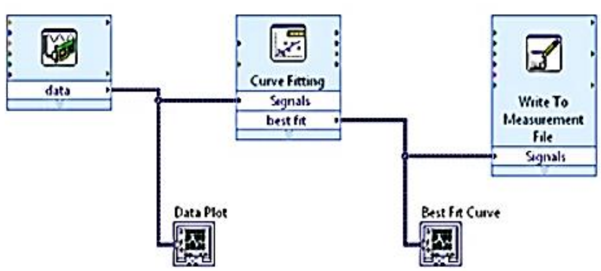

Fig. 3. Design of linear model optimization VI in LabVIEW

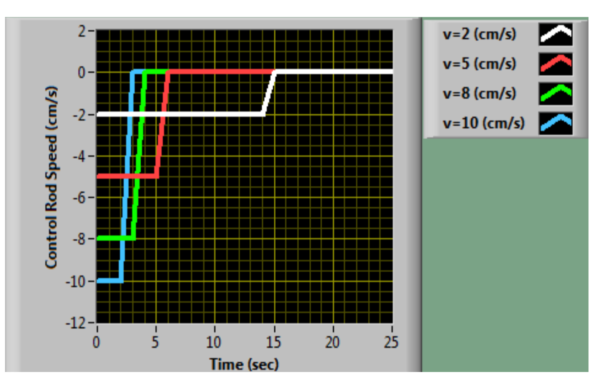

Fig. 4. Optimization of control rod speed with linear control rod reactivity model in LabVIEW

Amongst four design values of control rod speed, the desired objective is to obtain a maximum speed for safe operation of ACP1000 nuclear reactor. Therefore, the maximum safe control rod speed (MSCRS) is 2 cm/ sec which moves with optimal design time delay of 14 seconds. Now, it is desired that against maximum safe speed, the reciprocal of reactor period which is a measure of net reactivity under critical operation of the reactor must be zero as shown in Figure 5. This proves that reactor is self-regulating with internal and external reactivity feedbacks.

The behavior of maximum relative reactor power (MRRP) against four different control rod speed is shown in Figure 6. The maximum power level using ex-core instrumentation for large power excursion is 110% RP having equivalent relative reactor power of 1.1 on normalized or per unit scale. This value is basically the design safety limit before the actuation of protection system. The

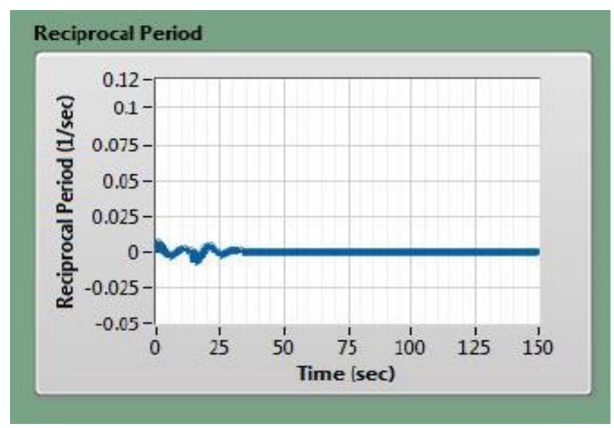

Fig. 5. Optimization of reciprocal reactor period at optimal control rod speed with linear control rod reactivity model in LabVIEW

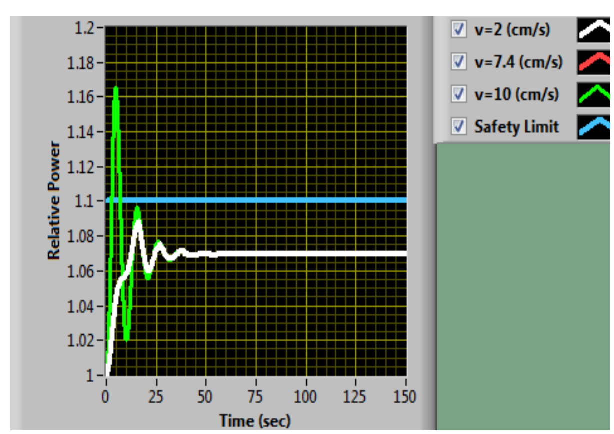

Fig. 6. Optimization of relative reactor power with linear control rod reactivity model in LabVIEW

safest power level is found as 107% RP which is well below the design safety limit (110%).

The optimal trend of average coolant temperature (ACT) is shown in Figure 7. The coolant cold leg temperature is 291.5 ºC, coolant hot leg temperature is 328.5 ºC, and initial reactor coolant average temperature is 307.8º C. The optimized value of average coolant temperature by using CORTH Code is 310 ºC with ± 2.8 ºC uncertainty. However, this average coolant temperature is overestimated by 2.2℃.

The optimized coolant average temperature using GNTHACP Code is 307.4 ºC which is an excellent estimate as compared to CORTH Code.

3.2 Modeling and Nonlinear Optimization in LabVIEW

In this research work, a VI is designed for nonlinear SQP optimization as shown in Figure 8. The design of cost function for nonlinear SQP optimization VI is shown in Figure 9, while the design of front panel for nonlinear SQP optimization VI is shown in Figure 10.

The VI is designed for the computation of neutronics and thermal hydraulics system model parameters as shown in Figure 11. A VI is designed to model neutronics and thermal hydraulics model equations as shown in Figure 12. A VI is designed to calibrate the neutronics and thermal hydraulics system model as shown in Figure 13.

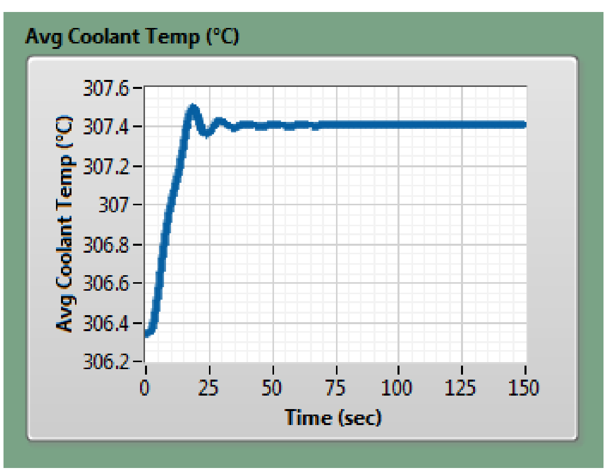

Fig. 7. Optimization of average coolant with linear control rod reactivity model in LabVIEW

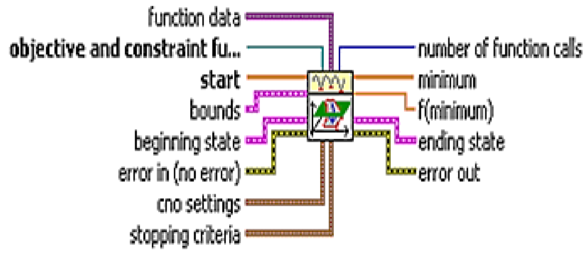

Fig. 8. Design of nonlinear SQP optimization VI in LabVIEW

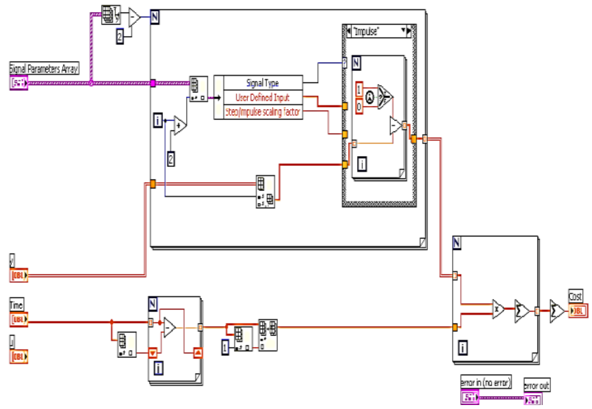

Fig. 9. Design of cost function for nonlinear SQP optimization VI in LabVIEW

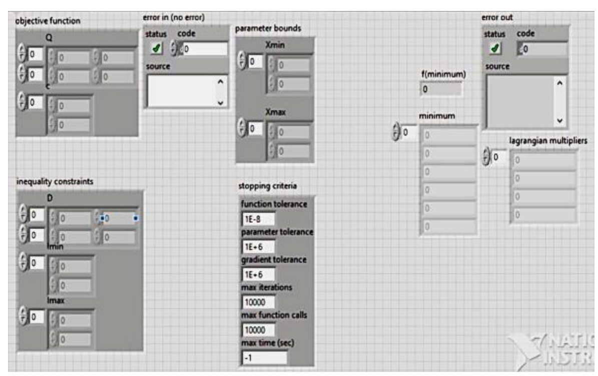

Fig. 10. Design of front panel for nonlinear SQP optimization VI in LabVIEW

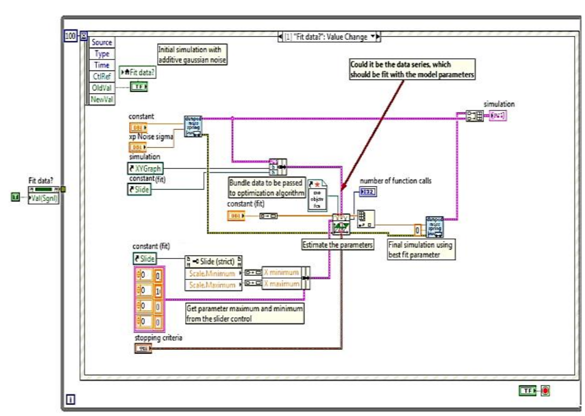

Fig. 11. Computation of neutronics and thermal hydraulics system model parameters in LabVIEW

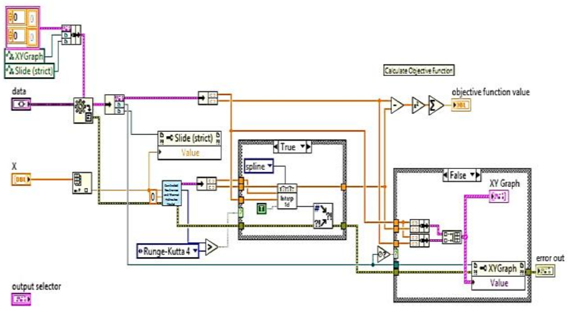

Fig. 12. Modeling of neutronics and thermal hydraulic system equations in LabVIEW

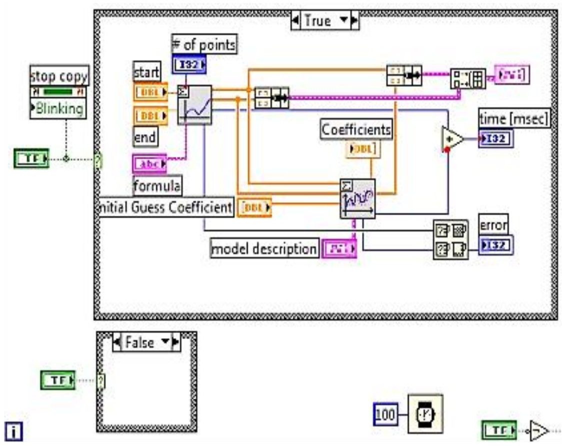

Fig. 13. Calibration of neutronics and thermal hydraulics system model in LabVIEW

The comparison of optimization parameters of both algorithms are tabulated in Table 1.

The behavior of relative reactor power against two different control rod speed is shown in Figure 14. The safest power level is found 107.33% RP which is well below the design safety limit. The optimal trend of average coolant temperature is shown in Figure 15.

The comparison of parameters of coupled neutronics and thermal hydraulics model of GNTHACP code and benchmark FSAR results computed

Table 1. Comparison of optimization parameters

GNTHACP Code	Design values
SIMPEX Algorithm Optimization Time (Sec)	70
SQP Algorithm Optimization Time (Sec)	52.5
Number of total gradient evaluations in SQP	31
SQP Lagrangian Multiplier	0.0036
SQP Penalty Factor	0.5
SQP Cost Function	0.0013

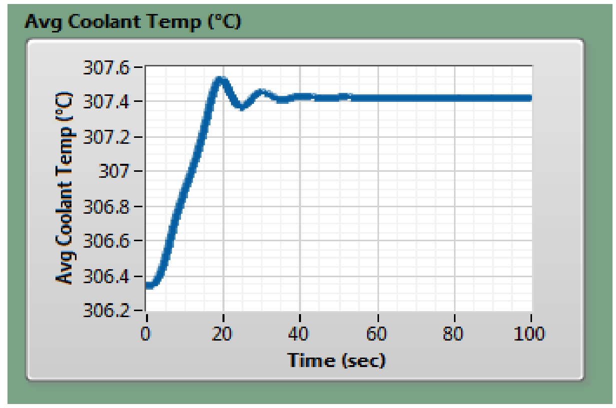

Fig. 14. Optimization of relative reactor power with nonlinear control rod reactivity model in LabVIEW

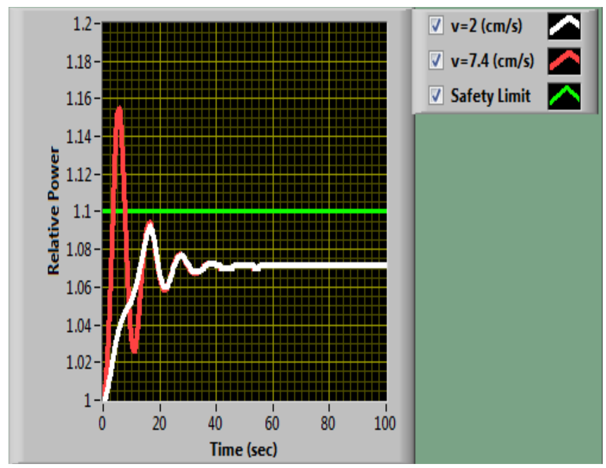

Fig. 15. Optimization of average coolant and fuel temperatures with nonlinear control rod reactivity model in LabVIEW

using CORCA and CORTH Codes is tabulated in Table 2. The results show that the proposed GNTHACP Code is quite accurate and hence a successful realization has been made.

4. CONCLUSION

The reactor neutronics and thermal hydraulics modeling has been successfully attempted and a state-of-the-art nuclear code (GNTHACP) is designed and developed in graphical programming environment LabVIEW. The GNTHACP nuclear code in LabVIEW is a step towards new toolkit development for the ACP1000 nuclear power plant neutronics and thermal hydraulics modeling in LabVIEW. The GNTHACP nuclear code is 100% equivalent to coupled CORCA and CORTH nuclear codes. The performance of GNTHACP nuclear code has been tested and validated against

Table 2. Parameters of coupled neutronics and thermal hydraulics model of GNTHACP code

Parameters	GNTHACP Code	FSAR Benchmark
MRRP (% RP)	107	110
MSCRS (cm/sec)	2	1.905
AFT (ºC)	653	650
ACT (ºC)	307.4	307.8

FSAR as benchmark and found robust. As such, the robustness of the GNTHACP nuclear code is established as tested and validated under extreme safety limits of ACP1000 imposed over the neutronics and thermal hydraulic parameters to ensure the design and optimization process valid under maximum allowed perturbing conditions. The results of control rod speed, reciprocal reactor period, relative reactor power, coolant temperature have been investigated and found industry standard toolset (IST) for neutronics and thermal hydraulics modeling of ACP1000 nuclear power plant. The proposed code development has established a strong basis for similar development for nuclear reactor systems other than ACP1000 in future.

5. ACKNOWLEDGEMENTS

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6. CONFLICT OF INTEREST

The authors declare no conflict of interest.

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