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STEADY STATE AND TRANSIENT THERMALHYDRAULIC ANALYSIS OF PHWR USING COBRA-3C/RERTR*

A. HUSSAIN[1**], F. ABOLABAN[2], S. M. KHUBAIB[3], S. MUBIN[4] AND I. AHMED[5]

[1,2]Dept. of Nuclear Engineering, King Abdulaziz University, Jeddah, KSA
Email: ahassain@kau.edu.sa
[3]Dept. of Mechanical Engineering, Nazeer Hussain University, Karachi, Pakistan
[4]Dept. of Civil Engineering, University of Engineering & Technology, Lahore, Pakistan
[5]Dept. of Chemical Engineering, University Technology PETRONAS, Perak, Malaysia

Abstract– Nuclear cross sections that determine core multiplication strongly depend on core temperature (e.g., the Doppler, moderator density effects etc). On the other hand, since this heat is generated by the neutron flux in the reactor core, the temperature distribution in the core will depend heavily on its neutronic behavior. Fuel centerline temperature could be the limiting constraint on reactor power because of the concern for fuel melting. Likewise, high clad temperature is also a possible limiting factor on reactor power because of the potential degradation of clad material or on-set of critical heat flux phenomenon.

An assessment of the steady state and transient thermal hydraulic capabilities of the computer code COBRA 3C/RERTR was made using model for a PHWRs reactor core. The temperature distributions determined for fuel, clad and coolant are compared with analytical results and with the results quoted in safety report. It was found that when the code was run for full power at reduced flow of 70% the bulk coolant temperature remained below the saturation temperature, so there is an adequate design margin is available for safety related scenarios.

Keywords– Thermal hydraulic analysis, PHWRs, fuel temperatures, COBRA/RERTR code, reactor safety

1. INTRODUCTION

The RERTR (Reduced Enrichment Research and Test Reactor) version of COBRA is based on the MIT version of COBRA-3C which performs thermal hydraulic nuclear reactor sub-channel analysis for research and test reactors. It was developed at Argonne National Laboratory, USA, in the early 80's. The purpose for modifying COBRA-3C / MIT was to make the code more suitable for research and test reactors which are operated at low pressure and temperatures. However, it can be used to analyze the sub-channel of a power reactor. The code has subroutines that can calculate the coolant properties for heavy water at various pressures and temperatures with reasonable accuracy. In addition, various sub-assemblies configuration can also be modeled in the said version of the code [1]. An assessment of the steady state and transient thermal hydraulic capabilities of the code was made, using model for a PHWR reactor core. Arshi et al (2010) have performed the thermal-hydraulic analysis of a typical VVER-1000 core at steady-state condition, using COBRA-EN code. Thermal-hydraulic analysis was done for the hottest channel and the results were compared with the Final Safety Analysis Report of the VVER-1000 reactor [2]. Aghaie et al (2012) have used an upgraded version of COBRA-EN for the analysis of palte type fuel element upgraded to include analysis of plate of the Tehran Research Reactor [3]. Natural convection model was used for estimation of hottest channel which show adequate match with safety analysis report of the

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**Corresponding author

research reactor. Gabriel et al (2011) demonstrated the use of sub-channel code F-COBRA-TF to determine the void fraction with reasonable accuracy [4].

The nuclear analysis of a reactor core is intimately related to its thermal analysis. Nuclear cross sections that determine core multiplication strongly depend on core temperature (e.g., the Doppler, moderator density effects etc.). On the other hand, since this heat is generated by the neutron flux in the reactor core, the temperature distribution in the core will depend heavily on its neutronic behavior. Fuel centerline temperature could be the limiting constraint on reactor power because of the concern for fuel melting. Likewise, high clad temperature is also a possible limiting factor on reactor power because of the potential degradation of clad material or on-set of critical heat flux phenomenon.

It should be mentioned here that COBRA is mostly used for thermal hydraulic analysis of PWRs. As we know, there are no papers available where COBRA has been used for PHWRs analysis. In this way it may serve as a novel study for using COBRA for PHWR analysis. The main purpose of the work presented here is to examine the effectiveness of RERTR for PHWR thermal hydraulics analysis. The code has been extended to include D_2O specific properties. Simulation results are validated with safety analysis report of CANDU reactor.

2. THERMAL HYDRAULIC ANALYSIS

In the single channel model, mass, momentum and energy exchanges between sub-channels are not considered. The coolant channel is considered as closed, isolated channel. But in the actual core, the lattices are open, where mass, momentum and energy exchanges between adjacent sub-channels must be considered.

Over the past few years, a major effort has been devoted to the development of techniques that would allow the analysis and prediction of the thermal-hydraulic behavior of reactor fuel assemblies [5, 6]. However, single channel analysis is commonly performed as the first approximation tool. The fuel heat transfer model considers radial conduction within the fuel by dividing the fuel into equally spaced concentric rings. The complete fuel is divided into N nodes of which N-1 are in the metal and one node is for the cladding. This gives N+1 temperatures Ti, where i = 1 is at rod center, i = N at rod surface and i = N+1 at the outer surface of the cladding (Fig. 1).

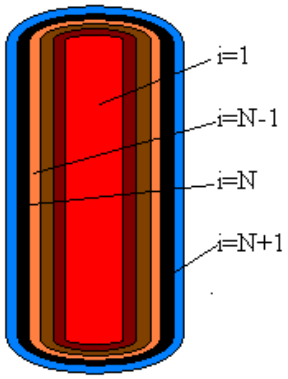

Fig. 1. Radial Nodal system for the simulation for an axial segment

The system steady-state calculation is of particular importance in preparing for the transient calculation. The model steady-state condition is adjusted to match the physical system's initial condition. The code contains a “steady-state” option in reaching the correct initial conditions. Since the steady-state condition represents the initial fluid conditions and the metal mass initial conditions, the “steady-state” option enables the user to quickly reach steady-state thermally and hydraulically by reducing the specific heats of

the metal masses to a low value. Thus, the model quickly converges to a condition representative of fluid conditions either input by the user or consistent with the user-input controllers. It is important to allow the model to run for a sufficient length of time before concluding that a steady-state condition has been reached. Once a steady-state condition has been reached, the fluid conditions should be virtually unchanging with time.

The reactor core is divided into a number of channels, considered as interacting in the case of a pin type reactor with cross flows and separated in a plate type reactor with no cross flows. The axial and radial power distribution in the reactor is calculated from the neutron flux and provided as input to the code. Conservation equations for mass, energy and momentum are solved to give the enthalpy and flow distribution throughout the reactor. The temperature profile in the fuel is calculated from the heat generation rate and the heat transfer coefficient between clad and coolant. From this information, the coolant temperature, coolant density and clad temperature are calculated.

The governing equations are coded in their transient form. The steady state conditions are obtained by setting terms d/dt to zero. The equations may be derived in differential form by considering the transport in and out of a control volume consisting of a length dx of channel i. To develop a better understanding of the basic equations, general sub-channel governing equations are discussed in the following sections.

Based on mass conversation, mass conservation equation of i^{th} axial section (Fig. 1) can be given by,

$$m_{j,i,in} - \sum_{r=1}^{Nj} \Delta m_{jk,i,r} = m_{j,i,ex},$$

Where

$m_{j,i,in}$ = Inlet mass flow rate of ith reaction in sub channel j,

$\Delta m_{jk,i,r}$ = Cross flow rate from sub channel j to sub channel k in the ith section,

$m_{j,i,ex}$ = Outlet mass flow rate of ith reaction in sub channel j, and

Nj = Number of sub channel adjacent to sub channel j

Based on heat balance, the energy out of the i^{th} section is equal to the energy entered into that section plus heat generated by fuel element minus heat transferred laterally by cross flow and turbulent mixing.

$$m_{j,i,ex}\, h_{j,i,ex} = m_{j,i,in}\, h_{j,i,in} + \bar{q}_{l,j,i}\Delta z - \sum_{r=1}^{Nj} Q_{jk,i,r} - \sum_{r=1}^{Nj} M_{jk,i,r}\,(h_{j,i} - h_{k,i})\,\Delta z$$

Where $h_{j,i,ex}$ = Fluid outlet enthalpy of ith section in sub channel j,

$h_{j,i,in}$ = Fluid inlet enthalpy of ith section in sub channel j,

$\bar{q}_{l,j,i}$ = Linear heat generation of ith section in sub channel j,

$M_{jk,i,r}$ = Mass flow rate due to turbulent mixing from sub channel j to sub channel k in the ith section,

Δz = Axial length of ith section of sub channel j

$\bar{h}_{ji}$ = Average coolant enthalpy of ith section in sub channel k,

$Q_{jk,i,r}$ = Heat transferred by cross flow due to pressure gradient between sub channel j and k,

$$\bar{h}_{j,i} = \frac{1}{2}(h_{j,i,in} + h_{j,i,ex})$$

$$\bar{h}_{k,i} = \frac{1}{2}(h_{k,i,in} + h_{k,i,ex})$$

$$Qjk,i=\frac{1}{2}\left[\left(hj,i-hk,i\right)mj,k+\left(hj,i+hk,i\right)mj,k\right]$$

$$Qjk=-Qkj$$

Axial momentum conversion equation of ith section in sub channel j, is,

$$\begin{aligned}\Delta Pj,i,ex &= P1-Pj,i,ex\\ &= \Delta Pj,i,in+\Delta Pj,i,+\Delta Pj,i,a\\ &+ \Delta Pj,i,ca,+\Delta Pj,i,el+\Delta Pj,i,gd\\ &+ \Delta Pj,i,ex,loc,\end{aligned}$$

where

P1 = Inlet pressure of reactor core,
ΔPj,i, ex = Pressure drop from core inlet to ith section outlet in sub channel j,
ΔPj,i,in = Pressure drop from core inlet to ith section inlet in sub channel j,
ΔPj,i,f = Friction Pressure drop of ith section in sub channel j,
ΔPj,i,a = Axial acceleration pressure drop of ith section in sub channel j,
ΔPj,i,el =Elevation pressure drop of ith section in sub channel j,
ΔPj,i,gd = Form pressure drop due to grid spacer of ith section in sub channel j,
ΔPj,i,ex, loc = Form pressure drop at outlet of ith section in sub channel j,
ΔPj,i,ca =Additional axial pressure drop due to cross flow between sub channels j and k in ith section.

$$\Delta Pj,i,ca=\begin{cases}-\dfrac{\dfrac{1}{Aj}\Delta mjk,iVj,i}{3600}, & \Delta mjk,i\geq 0\\ -\dfrac{\dfrac{1}{Aj}\Delta mjk,i(2Vj,i-Vk,i)}{3600}, & \Delta mjk,i\end{cases}$$

where Vj,i & Vk,i = Coolant velocities of sub channels j and k in the ith section, and
Aj = Cross sectional area of sub channel j,

The variation in hydraulic conditions among the various sub channels lead to differing axial pressure drop. Hence at any given axial level there will be pressure gradients leading to cross flow.

$$\begin{aligned}\Delta Pj,i,ex-\Delta Pk,i,ex &= \left|\left(P_1-Pj,i,ex\right)-\left(P_1-Pk,i,ex\right)\right|\\ &= \left|Pk,i,ex-Pj,i,ex\right|\\ &= -\varsigma\, j,k,i\ \frac{\left|Gjk,i\right|\ \Delta\ Gjk,i}{(\rho\ j,i,ex+\rho\ k,i,ex)\ \ x\ (3600)^2}\end{aligned}$$

where

ζjk,i = Cross flow resistance factor between sub channels j and k in ith section.
ΔGjk,i = Mass velocity between sub channels j and k in the ith section.
ρj,i,ex = Fluid density at outlet of ith section in sub channel j,
ρk,i,ex = Fluid density at outlet of ith section in sub channel k.

If this analysis is performed for the hottest channel and the results are well within the safety limits that would mean that the entire core would operate within the safety limits. Since the PHWR model which we have analyzed has Moderator Dump Shutdown System, it has the disadvantage of losing a "Heat Sink" in case of LOCAs.

Table 1 shows the important Thermal hydraulic data for a typical CANDU reactor.

Table 1. Thermal hydraulic data for CANDU reactor [5, 6]

QUANTITY	VALUE
Fuel pellet diameter	0.561 in
Clad wall thickness	0.0165 in
Fuel rod diameter	0.6 in
Channel flow area	2.957 in^2
Channel length	192 in
Fuel average temperature	2001.2 ^oF
Average sheath temperature	543.5 ^oF
Fuel thermal conductivity	1.45 Btu / hr ft ^oF
Fuel specific heat	0.0732 Btu /lb ^oF
Fuel density	630.7 lb / ft^3
Clad thermal conductivity	9.234 Btu /hr ft^oF
Clad specific heat	0.0766 Btu/lb ^oF
Clad density	409.3 lb / ft^3
Fuel to clad heat transfer coefficient	1580Btu/hr.$ft^2{}^o$ F
Reactor maximum heat flux	0.3435 MBtu / hr . ft^2
Average to maximum axial flux ratio	0.649
Extrapolated core height	195.84 in
Inlet temperature	474.3 F
Coolant density	0.845 g / cm^3
Average coolant velocity at centre	24.9 ft / s
Exit pressure	1551.3 psia

The first step was to determine the best grid size that will provide the most accurate answers to the coupled mass, momentum and energy equation without overly taxing the limited computer resources. This was done by running the simulation with various combinations of axial and radial nodes. It was found that at 80 axial nodes, energy error, fluid exit temperature and fuel average temperature are almost converged. Subsequently, radial nodes were increased from 2 to 20. Number of radial nodes has no effect on energy error, because energy added (calculated from average heat flux value) and energy out (which depends on axial heat flux shape) remained same. However, COBRA provides volume weighted fuel average temperature, therefore, by changing radial nodes from 2 to 5, 34.2 % increase was observed in fuel average temperature. Finally axial nodes were fixed at 80 and radial nodes at 20 where finer grid had minimal gain inaccuracy. All subsequent runs used 80x20 grids.

3. RESULTS AND DISCUSSION

Analytical calculations were performed to verify the results of COBRA. For this purpose equations were developed by using constant physical properties for coolant, gap, clad and fuel. All calculations were done for central channel at full power and flow conditions. Table 2 shows the comparison of analytical calculations with values calculated by the code using 80 axial nodes and 20 radial nodes. To assess the capabilities of the code, temperatures of coolant, clad and fuel as calculated by the code were compared with PHWR design values. Table 3 shows this comparison.

Table 2. COBRA results comparison with analytical calculation

QUANTITY	COBRA RESULTS	ANALYTICAL RESULTS
T fluid ,exit (F)	568.33	566.99
Tclad , max (F)	576.5 @ 153.6”	583.58 @ 147.61”
Tclad , exit (F)	568.4	568.3
Tsurface , max (F)	843.9 @ 105.6”	837.77 @ 105.15”
Tsurface , exit (F)	582.8	576.65
Tcentre , max (F	3367.3@ 96.24”	3399.7 @ 97”
Tcentre , exit (F)	711.1	655.63

Table 3. COBRA results comparison with Design Values

QUANTITY	COBRA Results	Design Values [4]	% Error
Coolant outlet temperature (^{o}F)	566.9	571.0	0.7
Clad average temperature (^{o}F)	542.5	543.5	0.18
Fuel surface temperature (^{o}F)	844.8	850.0	0.6
Fuel maximum temperature (^{o}F)	3354.0	3320.0	1.02
Fuel average temperature (^{o}F)	2094.2	2080.0	0.68

After steady state initialization of the code at full power and flow conditions, COBRA was run for full power at reduced flows to simulate a reduced flow condition. After steady state initialization of the code at full power and flow conditions, COBRA was run for full power at reduced flows. Figures 3-5 compare various temperatures at 100 % and 70 % flow. As can be seen, even at 70 % flow T_{fluid} is below saturation temperature. By reducing flow, the maximum locations of Tclad and Tfuelcenter were expected to shift towards channel outlet, and values of temperatures were expected to rise. As can be seen from Figs. 2, 3, and 4 the trend shown by COBRA was same as was expected analytically.

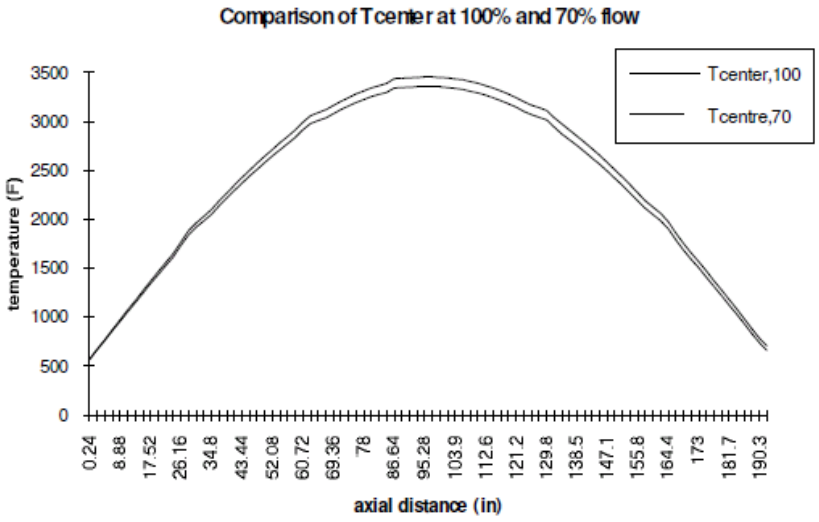

Fig. 2. Axial temperature profile comparison for 100% vs. 70% flow

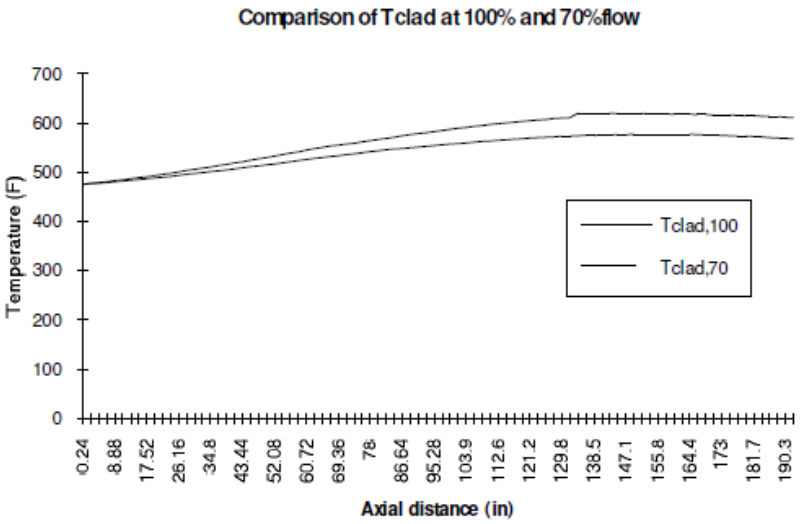

Fig. 3. Clad temperature comparison for 100% vs. 70% flow

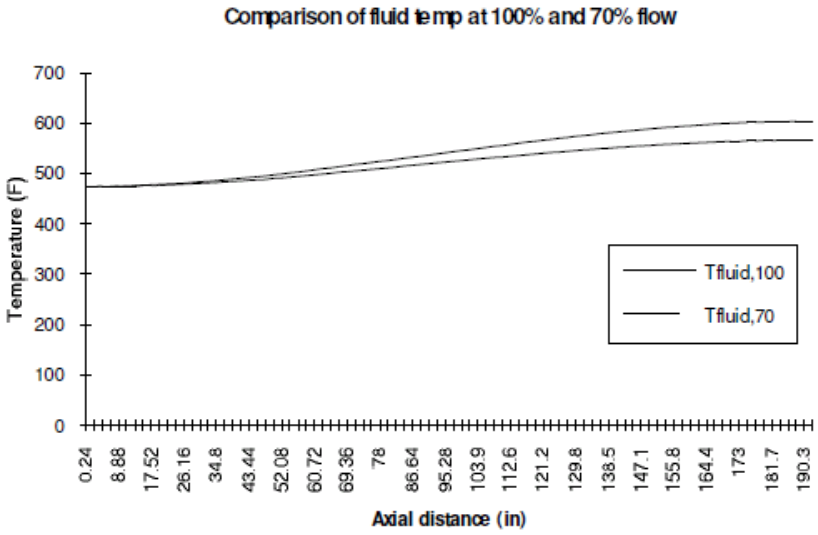

Fig. 4. Fluid temperature comparison for 100% vs. 70% flow

Subsequently, to assess the capabilities of the code for transient analysis additional simulations were performed. As the first step for the transient analysis, optimum time step was determined. The code was run for the PHT pump run down transient, using initial values of fuel thermal conductivity. Time step was changed from 0.5 second to 2.5 seconds. It was found that the effect of time step on transients is negligible as compared to the effect of the choice of an appropriate value of fuel thermal conductivity. Thus good approximations could be attained by using a crude time step, and by choosing average value of fuel properties. This could result in considerable reduction in running time of the code.

The code was run for a flow transient from 100 % flow to 90 % flow with varying time steps. The effect of time step is not so marked as compared to the effect of the total time of the transient. As total time of transient was increased, values converged to steady state values. It is logical because steady state values are at infinite time interval. The difference in T_{center} exit is due to the limitation of the code that it assumes constant fuel properties for all times. The effect of time step is given in Table 4.

Table 4. Effect of time step on converged values

Quantity	Steady State Values at values at 90% flow	Converged Transient		
		Total time =1 s Time step = 0.5 s	Total time =1 s Time step = 0.25 s	Total time =1 s Time step = 0.375 s
T_{fluid}, exit (°F)	576.66	572.52	572.25	576.14
T_{clad}, exit (°F)	549.4	547.7	547.6	549.1
T_{centre}, exit (°F)	2399.2	2394.4	2394.4	2394.4

Table 4 shows that the effect of time step is not so marked as compared to the effect of the total time of the transient. As total time of transient was increased, values converged to steady state values. It is logical because steady state values are at infinite time interval. The difference in Tcenter,exit is due to the limitation of the code that it assumes constant fuel properties for all times. As during the transient, fuel average temperature changes, therefore, fuel properties in transient were given at an average temperature of average fuel temperatures at 100 % and 90 % flow. In steady state at 90 % flow, kf=1.674 Btu / hr.ft.F but in transient kf=1.68 Btu / hr.ft.F. Due to this increased thermal conductivity, fuel temperatures were low during transient [7-10].

Also, the pump run down transient was simulated using the code. If a PHT pump trips, it supplies certain flow for approximately 50 seconds due to its inertia to remove decay heat which is being produced due to fission product decay. The primary heat transport system pump rundown curve matches decay heat as shown in Fig. 5. As can be seen from Table 5, the values for $T_{fluid,exit}$ and $T_{clad,mid}$ almost converged to steady state values. The error in $T_{fuel,mid}$ is again due to the constant value of fuel thermal conductivity for whole time span of the transient**.**

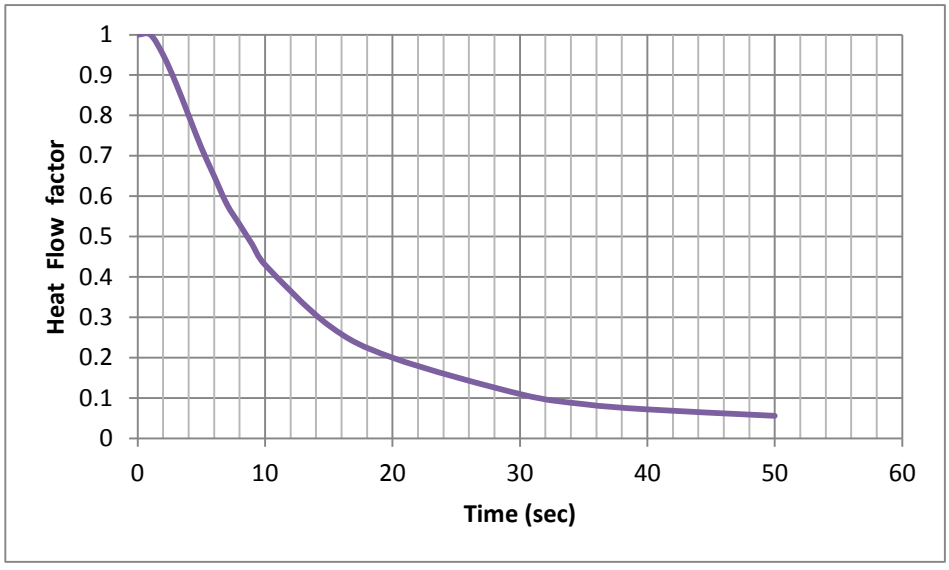

Fig. 5. Variation of decay heat as a function of time after reactor trip

Table 5. PHT pump run down transient with average value of fuel properties

Time (Sec)	T fluid, exit	T fuel, 96"	T clad, 96"	Steady State Values
0	566.96	2720.7	554.5	kf = 2.898 Btu / hr ft F T fluid, exit = 505.49°F T full, 96" = 596.4°F T clad, 96" = 497.2°F
5	588.94	2602.9	572.6	
10	591.07	2234.8	570.7	
15	584.68	1834.4	563.1	
20	570.22	1498.4	549.3	
25	557.68	1246.3	539.1	
30	548.2	1057.6	531.3	
35	538.61	916.6	522.1	
40	531.67	816.6	516.4	
45	526.83	748.3	512.2	
50	521.93	700.6	507.6	
55	516.82	668.8	503.5	
60	512.75	649.3	501.0	
65	510.04	637.3	499.5	
70	508.33	630.0	498.5	
75	507.25	625.5	498.0	

4. CONCLUSION

Initial comparison with the analytical solution suggested that the code is performing reasonably well in predicting maximum and exit temperatures in the channel. Therefore, for steady state analysis modified COBRA code is capable of predicting thermal hydraulic behavior of a PHWR. Various temperatures in the channel were also compared with the design values and the COBRA values are within a percent of the design value further validating the effectiveness of the code for heavy water power reactor modeling.

To simulate an off-normal situation of reduced coolant flow, a comparison was made between 100% and 70% flow through the channel. As can be seen, even at 70 % flow, T_{fluid} is below saturation temperature. By reducing flow, the maximum locations of T_{clad} and $T_{fuel,center}$ were expected to shift towards the channel outlet, and values of temperatures were also expected to rise. As can be seen from Figs. 2, 3, and 4, the trend shown by COBRA was the same as was expected analytically. Transient analysis results were also consistent with the expected results and it can be concluded that COBRA is suitable for PHWRs analysis

REFERENCES

1. Basile, D., et al. (1999). *COBRA-EN, an upgraded version of the COBRA-3C/MIT code for thermal-hydraulic transient analysis of light water reactor fuel assemblies and cores*. Radiation Safety Information Computational Center, Oak Ridge National Lab.
2. Safaei Arshi, S., Mirvakili, S. M. & Faghihi, F. (2010) Modified COBRA-EN code to investigate thermal-hydraulic analysis of the IranianVVER-1000 core. *Progress in Nuclear Energy*, Vol. 52, pp. 589–595.
3. Aghaie, M., Zolfaghari, A., Minuchehr, A., Shirani, A. & Norouzi, A. (2012) Transient analysis of break below the grid in Tehran research reactor using the newly enhanced COBRA-EN code. *Annals of Nuclear Energy*, Vol. 49, pp. 1–11

4. Gabriel, H. & Glück, M. (2011). Prediction of void fraction for PWR and BWR conditions with the sub-channel code F-COBRA-TF. *Nuclear Engineering and Design*, Vol. 241, pp. 3952– 3966.
5. PHWRs Final Safety Report (KFSR).
6. Siddiqui, M. S., Chishti, S. A. & Iqleem, J. (1992). *Thermal analysis computer programme package for the estimation of KANUPP coolant channel flows and outlet header temperature distribution*. Foreign Technology-Pinstech, Islamabad, Pakistan.
7. Aslam, M., Ali, S. & Akhtar, T. M. (1991). *KANUPP test simulator, Ver 2.0. reduced scope basic principle simulator of KANUPP*. Foreign Technology-PINSTECH, Islamabad, Pakistan.
8. Goodarzi, S. & Mortazavi, S. (2012). Numerical simulation of a buoyant suspending drop in plane couette flow: the equilibrium position of the drop. *Iranian Journal of Science & Technology, Transactions of Mechanical Engineering*, Vol. 36, No. M1, pp. 69-82.
9. Gupta, S. & Tewari, P. C. (2011). Maintenance performance evaluation of power generation system of a thermal power plant, *Iranian Journal of Science & Technology, Transactions of Mechanical Engineering*, Vol. 35, No. M1, pp. 47-59.
10. Turgut, O. (2014). Numerical investigation of laminar flow and heat transfer in hexagonal ducts under isothermal and constant heat flux boundary conditions. *Iranian Journal of Science & Technology, Transactions of Mechanical Engineering*, Vol. 38, No. M1, pp. 45-56.