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Flow blockage accident or loss of flow accident by using comparative approach of NK/TH coupling codes and RELAP5 code

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ABSTRACT

This paper deals with the comparison and safety studies of the neutron kinetics and thermal hydraulic coupling analysis of small nuclear reactor core. In this study, first the coupling between neutron kinetics and thermal hydraulics has been achieved for the small nuclear reactor core. The coupling approach is done by using the lattice physics code (HELIOS), neutron kinetic code (REMARK) and thermal hydraulic code (THEATRe). After finalizing the NK/TH coupling, power distribution of reactor core has been acquired. The loss of flow accident or coolant blockage accident is considered for the safety analysis of the reactor core. The comparison studies were accomplished by comparing the results from coupled codes (HELIOS, REMARK & THEATRe) and thermal hydraulic system code RELAP5. The simulated results from both codes agree well conforming the accuracy of the NK/TH coupling approach. The purpose of the research is to corroborate the NK/TH coupling by safety pre-assessment of nuclear reactor core.

1. Introduction

Safety analysis of the nuclear reactors has been analyzed by using a verity of different conservative and simplified models which could not give the actual phenomenon occurring inside the nuclear reactor core. Today, due to the increase in modern technologies like advanced computational facilities, sorted out the problem arising from the use of the thermal hydraulic system model. The thermal hydraulic model uses point or one dimensional kinetic model which is not giving the actual picture of the transient phenomenon in the reactor core.

The modern computational techniques include higher processor speed computers in which nuclear codes work perfect. Therefore, the trends of utilizing high speed computers and three dimensional analysis are enhancing correspondingly. The benefits and the applicability include the designing of nuclear reactor core, accident analysis, safety assessment, and predicting extending fuel cycle etc. (IAEA Safety report No. 29, 2003).

Nowadays, the trends of nuclear reactor simulation codes are developing across the globe and many countries and nuclear organizations are concerned in the development of such codes. The notion of coupling the nuclear codes is very useful especially when we are dealing with neutron kinetics (NK) and thermal hydraulic (TH) analysis of the nuclear reactor core. The coupling between NK/TH appraises the thermal feedback effects which are used to simulate the accident scenarios. This technique implicates the core spatial power distribution and the feedback effect between the two phenomena. Many accidents can be simulated involving symmetric core spatial power distribution and strong feedback effects between neutronics and thermal hydraulic. The postulated accidents that can be simulated by coupling codes techniques include loss of coolant accident, mean steam line break accident, control rod withdrawal accident, coolant blockage accident, and loss of flow accident etc., because during the accidents deformation in the radial and axial power distribution may occur. The coupling process is based upon the time exchange of neutronics to thermal hydraulic properties. The thermal hydraulic model employs power as heat source for conduction calculations while the neutronic models involve coolant/moderator and fuel properties to update the macroscopic cross section based upon the local node conditions. It also computes three dimensional fluxes and power distribution and sends node wise power distribution to the thermal hydraulic model.

Safety of nuclear reactors has been started since the beginning of the first nuclear reactor i.e. "Fermi Pile" CP-1 (Chicago Pile 1) in 1942 which was primarily employed in Plutonium (Pu) production. In this reactor, fast shutdown was driven by the operator by cutting the retaining rope with an axe. While the secondary shutdown had been made by emptied a bucket containing cadmium sulfate solution into the core when accident occurs. Afterward, many reactors have been built used for both military and civil purposes along with safety criteria (Review guide, 2008).

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Nomenclature

NK/TH	neutron kinetic/thermal hydraulic
INEEL	Idaho National Engineering and Environmental Laboratory
LWR	Light Water Reactor
PWR	pressurized water reactor
HELIOS	lattice physics code
REMARK	neutron kinetic code
THEATRe	thermal hydraulic code
RELAP5	thermal hydraulic system code
OTSG	once through steam generator
RPV	reactor pressure vessel
PVM	parallel virtual machine environment
AURORA and ZENITH	input/output processing codes in HELIOS
HERMIS	subroutine in HELIOS code
Keff	effective neutron multiplication factor
Ag	silver
In	indium
Cd	cadmium

Today, nuclear power plant incorporates both active and passive safety features. Since most reactor emergency cooling systems are active, the reactor has passive limitations of a power excursion through a negative power coefficient of reactivity. Passive and intrinsic safety solutions were adopted as being effective and economically convenient. Whereas, the fundamental safety features are narrowed to reactor shutdown, containment cooling and containment of radio toxic products (Gianni, 2006).

After the TMI and Chernobyl accident, the design engineers and operators were persuaded that accident prevention and mitigation in nuclear plants deserved serious attention upon multiple failure in the complex safety system and serious consequences of human error (Qing et al., 2009).

Afterward, attention was focused on passive safety and inherent safety systems. Passive system deals with the structural and designing events without relying on devices but have some drawbacks like less powerful and slower in action than their active control part. Inherent safety deals with the elimination of hazards of choice of material or design concept.

- The flow blockage accident has been construed by many scientists and researcher across the globe, some of the recent work done are given as;
- The analysis of the flow path blockage accident has been investigated by Lei Yang et al. in which cased assembly was analyzed by using RELAP5/MOD3.2 code (Yang et al., 2012).
- Ki Yong Choi et al. have employed VISTA test facility on SMART-P integral type nuclear reactor to investigate the different transient analysis (Choi et al., 2006).
- The flow blockage accident was analyzed by Liu Tian-cai et al. in the China Advance Research Reactor (CAAR) by using RELAP5/MOD3.2 (Tian et al., 2006).
- The total blockage in a single fuel assembly was analyzed by Pierro et al. by using RELAP5/MOD3.3 code (Pierro et al., 2004).
- CFD studies of MTR reactor has been done by Salama et al., by considering the accidents which are mainly the flow blockage through coolant channel (Salama and Morshedy., 2012a), flow blockage under loss of flow transient i.e. hot channel scenario (Salama and Morshedy, 2012b), and flow blockage under loss of flow transients on average channel scenario (Salama, 2012).

In the current research, small nuclear reactor core has been designed by coupling three codes i.e., HELIOS, REMARK and THEATRe. The radial and axial power distribution of the reactor core has been achieved and is comprehended in this paper. The focus of the research is to find out the safety analysis of the reactor core and in this case we simulate the loss of flow accident or coolant blockage accident. This accident is simulated by coupling technique and is verified by using thermal hydraulic system code RELAP5. The results obtained from both analyses are almost same, thus conforming the accuracy of the simulation.

In our research, we have deemed a small nuclear reactor with power output of 220 MW designed by the college of nuclear science and technology, Harbin Engineering University. PR China. The main design parameters of the reactor are given in Table 1.

Table 1
Design parameters of IPWR.

Core parameters	Designing values
Core power	220 MW
Fuel type	Plate type
Fuel used	$Zr_2 + UO_2$(Nb)25%
No.of fuel assemblies	55
No. of fuel plates in one assembly	$3 \times 20 = 60$
Total no. of fuel plates	3300
Number of control rod groups	6
Height of each fuel assembly	1.5 m
Fuel meat width	1.2 mm
Cladding thickness	0.4 mm
Total heat transfer area of core	937.431 m^2
Single channel flow width	2 mm
Circulation area of core	0.6562017 m^2
Primary coolant pressure	15.5 MPa
Core inlet/outlet temperature	558/597 K
Primary coolant flow rate	1004.3 kg/s
Total no. of pumps	6
OTSG's parameters	*Designing values*
Number of OTSG's	12
Primary tube diameter	14 mm
Secondary tube diameter	8 mm
No. of tubes	721
Height of tube	1.5 m
Lower plenum radius	0.27 m
Upper plenum radius	0.27 m
Steam pressure	3 MPa
Steam flow	81.6 Kg/s
Steam temperature	549 K
Water pressure	4.2 MPa
Water temperature	328.15 K
Core material	
Nuclear fuel	UO_2 + Nb(25%)
Coolant	Light water
Clad material	Zircaloy
Moderator	Light water
Control material	Ag–In–Gd(50% + 25% + 25%)
Core thermal hydraulics	
Fuel thermal conductivity (W/m K)	12.098
Cladding thermal conductivity (W/m K)	16.12
Fuel specific heat (kJ/kg K)	0.334
Cladding specific heat (kJ/kg K)	14.16
Fuel density (kg/m^3)	7.303
Clad density (kg/m^3)	7.0
Axial peaking factor	1.0
Inlet coolant temperature (K)	558
Outlet coolant temperature (K)	597
Operating pressure (MPa)	15.5

2. Neutron kinetics and thermal hydraulic coupling (NK/TH) analysis and methods

The research was conducted by coupling neutron kinetics and thermal hydraulic by employing computational technique. There are two perceptible approaches for the NK/TH coupling i.e., (Rademer et al., 2004; Shimjith et al., 2008; Khan et al., 2011a; 2012).

- Serial integration coupling.
- Parallel processing coupling.

In serial integration coupling, the source code of the NK code is to be modified by changing the subroutine of the code and then coupled with TH code. In this coupling approach, one code is set static while the other keeps running. NK code gives power distribution by collecting the cross-sectional data from lattice physic code at certain times. This power distribution is then acting as input to the TH code which was static and then running the code at the same time as NK code. The parameters like fuel temperature, coolant temperature, moderator temperature, void fraction, density etc., extracted from the TH code fed into the table interface which acts to generate new cross section and are built in NK code for generating new power distribution. This process relies upon the time step and will continue as presented in Fig. 1.

The parallel coupling approach is the one in which the two codes run simultaneously and exchanges the data at every time step with the help of a parallel virtual machine (PVM) as elucidated in Fig. 2.

In the current research, neutron kinetic and thermal hydraulic coupling have been carried out by using HELIOS, REMARK and THEATRe codes as shown in Figs. 2 and 3 (Helios Methods Manual Book, 2008; REMARK modeling techniques handbook, 2005a, 2005b; THEATRE user guide, 2005; Modeling Techniques for THEATRe, 2005).

The coupling process is such that the lattice physics code i.e., HELIOS builds the core geometry and takes some initial values to produce absorption cross section, fission cross section etc., which is then fed into the REMARK code which gives the power distribution of the reactor core in three dimensions.

This power distribution is used as input values for THEATRe code which gives the fuel temperature, coolant temperature, moderator temperature, void fraction, coolant density etc. The values acquired from the THEATRe code are used by REMARK code to generate new cross section and this process is then continued.

3. Computational techniques

3.1. HELIOS

HELIOS is two dimensional (2D) lattice physics high order transport code and can be able to solve any kind of geometry. This code employs neutron and gamma groups for the burn up calculations, flux distribution and micro/macro cross-sectional data. It works via two separate codes AURORA and ZENITH which are input and output processing codes. The data exchanges between the two codes are accessed by the subroutine package HERMIS (Helios Methods Manual Book, 2008).

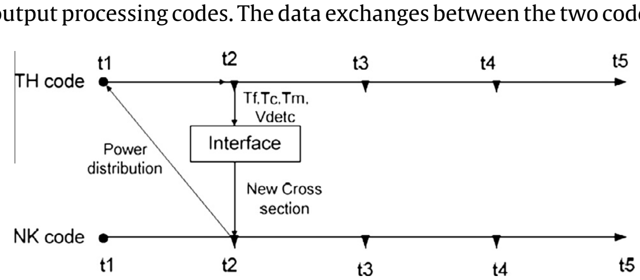

Fig. 1. Serial integration.

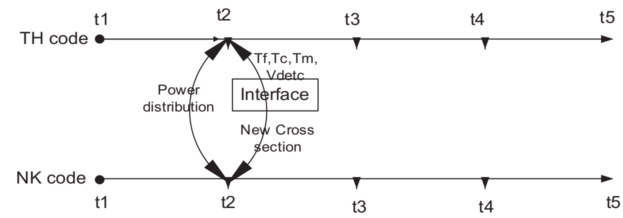

Fig. 2. Parallel processing coupling.

3.2. REMARK

REMARK (REal Time Multigroup Advanced Reactor Kinetics) code is developed by the GSE power system which is used for the real time simulation of nuclear reactor core. This code utilizes two groups, 3D time dependent diffusion theory in the form of finite difference equations to simulate nuclear reactor core. REMARK has two neutron energy groups to accurately simulate the characteristics of fast and thermal neutron under normal and abnormal operating conditions of the nuclear reactor. It can be modeled the core geometry as 3D mesh cell structure under the limited capacity of computer resources for real time applications. Mesh cell sizes are elected to ensure that the material properties within the cell are homogenous or heterogeneous depends upon the fuel, reflector, shielding etc. In the REMARK code, there is improved quasi-static solution for obtaining the flux distribution and six delayed neutrons to be calculated in each mesh cell. Reactivity feedback is based upon the core thermal hydraulic conditions like fuel temperature, coolant temperature, moderator density and void fraction. This code can calculate the reactivity effect of control rod movement and soluble boron. It can also give the exact neutron flux readings at in-core and ex-core detector points and utilizing neutron cross section for neutronic calculations (REMARK modeling techniques handbook, 2005a, 2005b).

3.3. THEATRe

This code is derived from the well-known thermal hydraulic system code RELAP5, so in this criteria, a part of the RELAP5 methodology has been merged with the real time simulation methodology. The selected methodology comprises field equations, the interfacial mass balance equation, the state equations, flow regime maps, constitutive correlations and solution method (THEATRE user guide, 2005; Modeling Techniques for THEATRe, 2005).

THEATRe code works via two steps, the first step corresponds to major computational techniques leading to solve thirteen linearized equations per fluid node. These equations have been solved for thirteen variables which are nodal pressure, the phasic equilibrium temperature, the non-condensable gas concentration, the phasic mass transfer, the void fraction, the phasic internal energies, the phasic densities, saturation temperature at specified pressure and the phasic velocities. Thirteen variables require thirteen independent equations which are as follows:

- Non-condensable gas mass conservation equation (1).
- Phasic mass conservation equation (2).
- Mixture momentum balance equation (1).
- Drift flux equation (1).
- Phasic energy conversion equation (2).
- Equation of state for phasic temperatures (2).
- Equation of state for phasic densities (2).
- Equation of state for saturated pressure (1).
- Interfacial mass balanced equation (1).

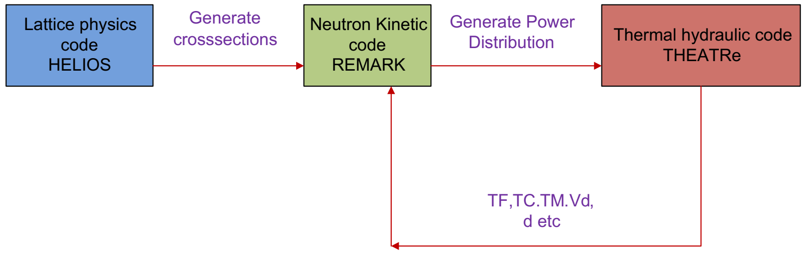

Fig. 3. Simulation method NK/TH coupling.

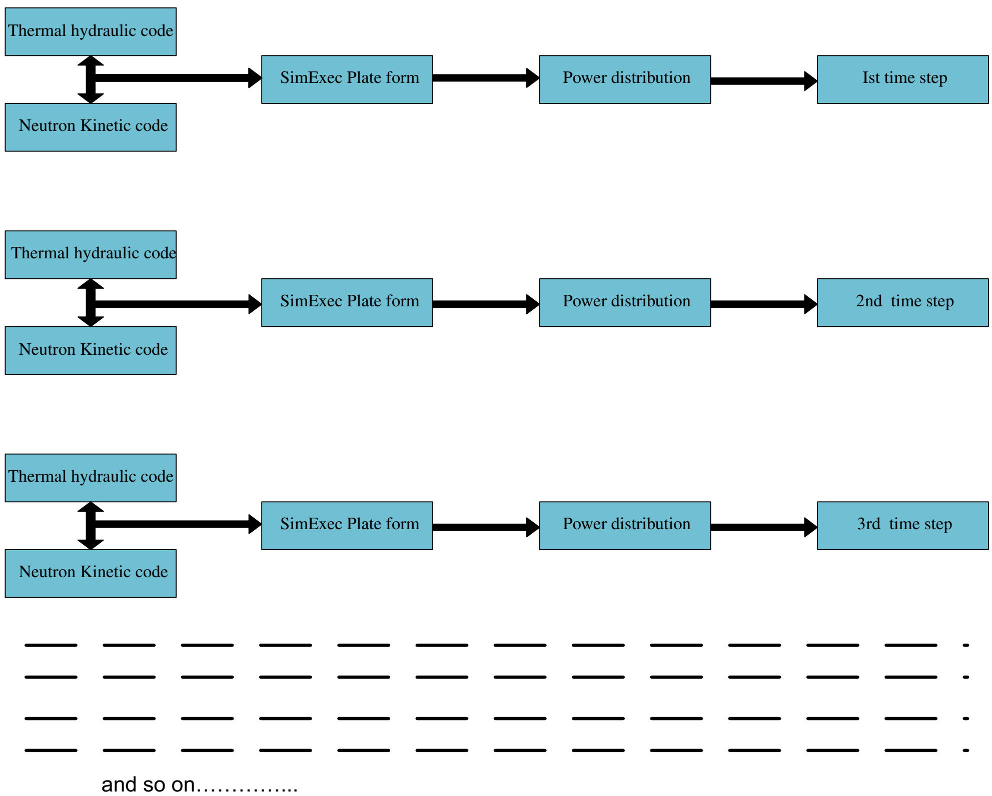

Fig. 4. Number of iteration steps.

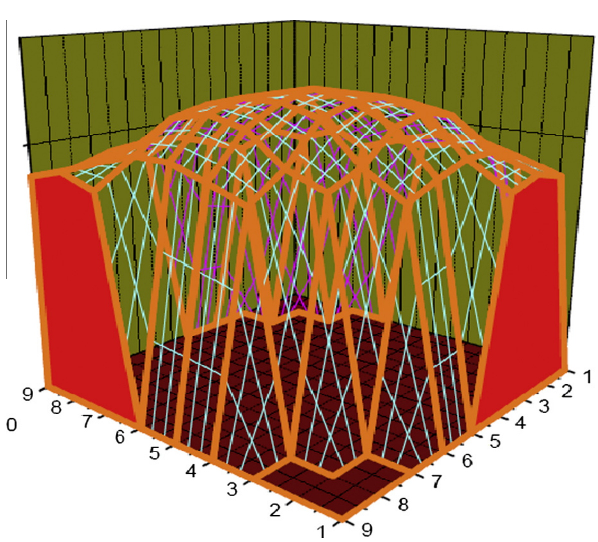

Fig. 5. 3D power distribution of core.

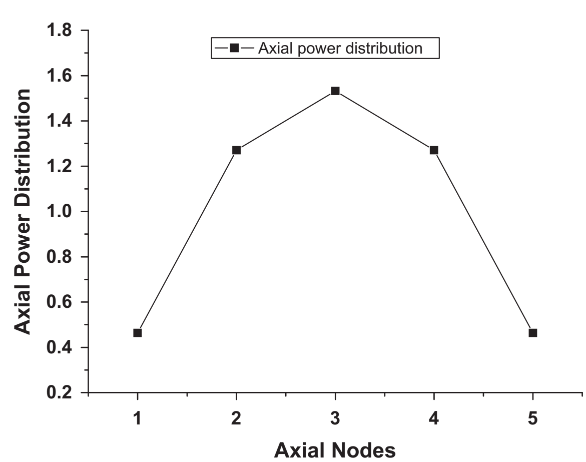

Fig. 6. Axial power distribution of the reactor core.

The first step is used the non-conservation form of the field equation, while the second step is used to augment the mass and energy conservation i.e., to minimize mass and energy errors.

3.4. RELAP5 code

RELAP5 code is a thermal hydraulic system code developed by the Idaho National Engineering and Environmental Laboratory (INEEL). Several versions of the RELAP5/Mpd3.4 code have been developed by INEEL and the specific applications of the codes include simulation of Light Water Reactor (LWR) systems such as coolant blockage accident (Martina et al., 2005; Wenxi et al., 2007), Loss of flow accident, Loss of Coolant Accidents (LOCAs), Anticipated Transient Without Scram (ATWS) and also operational transients such as Loss of Feed Water (LOFW) (RELAP5/MOD3.3 Code Manual, 2001a, 2001b; Khan et al., 2011b, 2013; Khan and Peng, 2011), loss of off-site power, mean steam line break accident (Khan et al., 2011c), station blackout and turbine trips type accidents etc.

4. Research methodology

In this research, we have designed a small nuclear reactor core by using the coupling approach techniques (NK/TH). The coupling process is presented in Fig. 3, here, we are not mentioning the detail studies of the reactor core design (Khan et al., 2011a) instead, we simulate the safety assessment of the reactor by using a

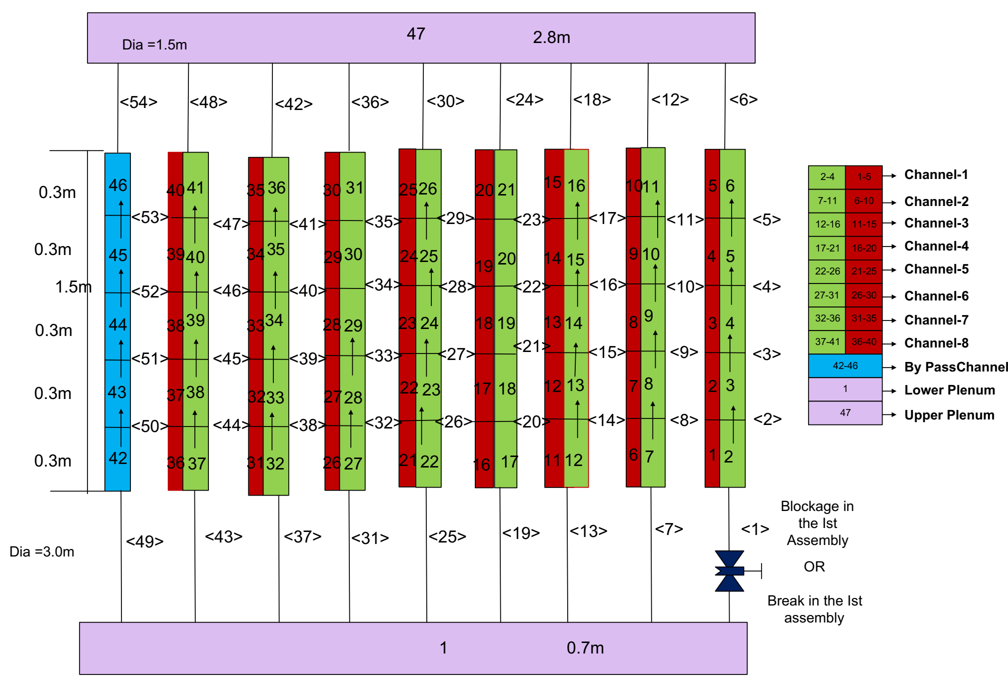

Fig. 7. Flow blockage and LOFA accident scenario.

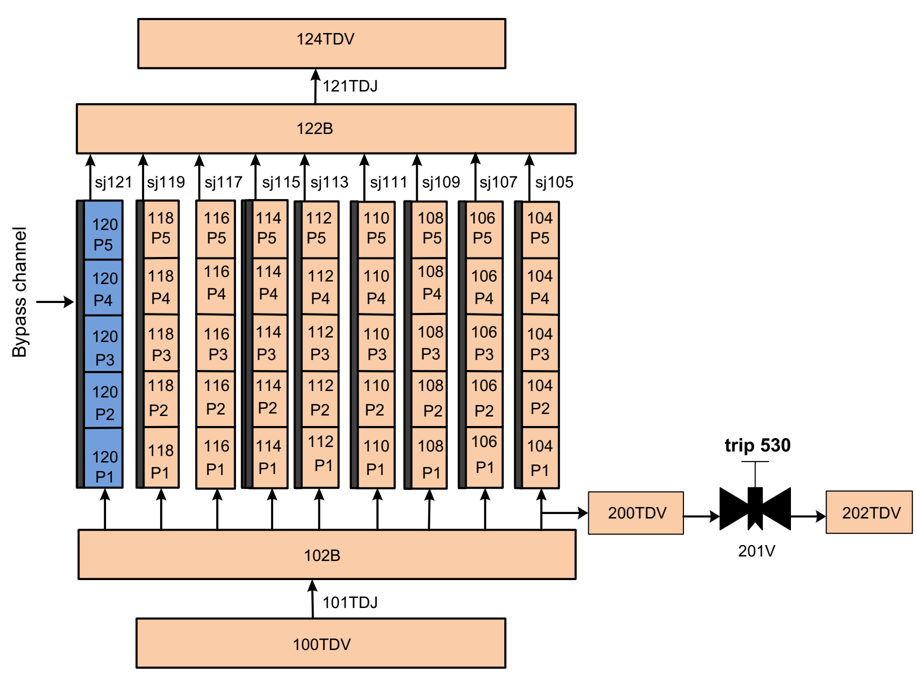

Fig. 8. Flow blockage and LOFA accident scenario simulated in RELAP5 code.

comparative approach of coupling code technique and RELAP5 code. The results procured from both analyses give the comparison of the research being conducted.

In this paper, we analyze the coolant flow blockage accident and the loss of flow accident by coupling approach of the same nuclear reactor core.

In order to carry out the reactor core design studies, we have considered the basic design parameters of the reactor as listed in Table 1.

Since our chief concern is with the three dimensional NK/TH coupling, so we have just computed some parameters which are required for the coupling process. In addition to this, we have checked out the criticality of the reactor which conform the core design (Khan et al., 2011a; Baturin, 2011; Bruna et al., 2007; Trevor et al., 2008).

After finalizing the core design studies, we have generated some cross sections and then used to input in the REMARK code for the evaluation of the 3D power distributions (Khan et al., 2011a; Li et al., 2008; Daniel and Thomson, 2003).

For the NK/TH coupling process to be accomplished, the power distribution of the reactor core should be radial or axial and that means NK/TH coupling has been obtained. While if we do not acquire the power distribution radial or axial, then it depicts that the NK/TH coupling has not been achieved. In our research studies, we got the radial power distribution representing the NK/TH coupling to be achieved successfully. In this case, we do not need for other iteration steps (Fig. 4) as this will give the same power distribution again (Khan et al., 2011a).

4.1. Number of iteration steps

One iteration step corresponds to the coupling of the two codes running simultaneously to get the power distribution as presented in Fig. 4. In the next time step, modified cross section is automatically sent to neutronic code to generate new power distribution. In the same time, thermal hydraulic code adjusts automatically to update the parameters like fuel temperature, coolant temperature, void fraction, moderator temperature etc. This whole parameter transfer is accompanied by using SimExec plate (PVM) form which acts as an interface between the two codes. The SimExec plate form is required to exchange the data between the codes simultaneously.

4.2. Boundary conditions

The fuel temperature, coolant temperature, moderator temperature, void fraction etc., are the boundary conditions for the coupling process. These parameters have been used in the HELIOS code for generating cross section and also used in the THEATRe code in the coupling process to be achieved.

In our analysis, we got the power distribution of the reactor core by coupling approach as depicted in Figs. 5 and 6. The data obtained from the THEATRe code were then applied to the table interface for generating the new cross section. This new cross section will then give the power distribution which is portrayed in Figs. 5 and 6 representing axial and radial power distribution respectively.

5. Safety assessment analysis

After finalizing the research studies of the reactor core, some accident scenarios were considered for the safe evaluation of this reactor core. For this purpose, the studies have been performed for the coolant blockage accident in an assembly and loss of flow accident.

5.1. Coolant flow blockage accident and loss of flow accident

The coolant flow blockage accident is not an ordinary accident in the history of nuclear power plant which merely can occur. Go through the history, in the year 1975, blockage in the coolant inlet flow was happened in the BR2 reactor of the Belgian Nuclear Research Centre (SCK•CEN). In this accident, the two plates of the reactor core had been melted resulted into the blockage of the flow channel. This happens during the fuel loading process a foreign object i.e. plastic handles of screw driver had dropped in the fuel channel (Martina et al., 2005; Wenxi et al., 2007) resulting into the blockage of flow channel due to the meltdown of the fuel plates.

Afterward, much research on the analysis of partial and full blockage of assembly channels has been undertaken on IAEA 10 MW MTR pool type research reactor.

The loss of flow accident can be happened due to primary pump failure which comprises loss of pump energy supply, fuel channel blockage, the primary coolant flow reduction due to valve failure, blockage in piping or in heat exchangers. The results of these accidents lead to the reduction in the flow rate in core channel.

5.1.1. Simulation of LOFA accident with coupling approach

For the evaluation of the flow blockage accident, we made the assumption that in the first assembly of the reactor core, and there appears flow blockage due to any unforeseen reason such as the swelling up of the fuel plates which then leads to decreases the flow area of the coolant or because of some object that has been falling during installation of the assemblies in the reactor core (Yang et al., 2012).

For the simulation of the accident, we have connected a valve in the inlet of the channel representing the flow blockage accident scenario. This accident can also be called as the loss of flow accident as nearly the same assumption should be considered for the desired accident.

Here, the important thing is that the accident happens in the first assembly of the reactor which lies at the center of the reactor core as shown in Fig. 7 (represents the nodalization diagram for the accident scenario). The first channel represents one assembly, the second, third, fourth, sixth and eight assembly contains same number assemblies i.e. 6 while, channel 5 and channel 7 contains the same number of assemblies i.e. 12. The ninth channel represents the bypass channel.

It is significant to consider the interaction between the reactor's cooling loop and the reactor core's kinetics during the flow block-

Table 2
Sequence table for accident analysis.

Time sequence of events (s)	Event description
0–10	Normal operation
10	Partial blockage occurs
10 + 495	Steady state value pertaining to blockage accident
500	End of simulation

Table 3
Boundary conditions for accident analysis.

Key parameters	Partial blockage
Power	0.9545 MW
Blockage position	9545%
Blockage course time	10 s
Scram	Disable

age or loss of flow accident scenarios. The fuel heat conduction and fuel coolant heat convection model should be considered in this analysis.

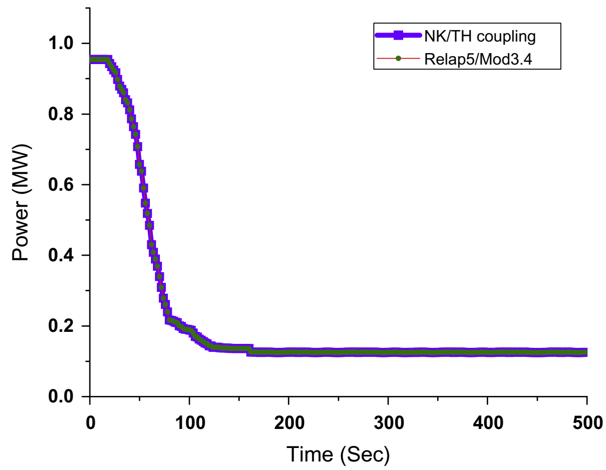

Fig. 9. Reactor power trends in NK/TH coupling and RELAP5 code.

The flow blockage accident is simulated such that at 10 s the accidents happen and it lasted up to 500 s. The time sequence of events and the boundary conditions for the accident are included in Tables 2 and 3.

The effect of different parameters under the time sequence of the accident is presented in Figs. 9–17.

In Fig. 9, the normal power level of the channel is shown. It can be seen that with the beginning of the accident the power goes down of the effected assembly as was explain earlier.

In Fig. 10, it is shown that the temperature of the fuel has varying effects, at the start of the accident the fuel temperature was 613 K which is the normal operational temperature. After 10 s of the normal operation, there is a sudden blockage of the flow channel which decreases the mass flow rate of the coolant across the fuel assembly. This will then increase the temperature of the fuel to the alarming value of about 678 K which is less than the melting temperature of the fuel plates. So, it can be said that although the accident happened but it has not melted the fuel plates. Since, the melting point of fuel plate is very high. It can be observed in Fig. 10 that with the variation in time step the fuel temperature has unstable effect which leads to the conclusion that then there is still some grounds for the coolant to flow across the channel.

In Fig. 11, it shows that since the coolant normal temperature was 558 K but with the start of the accident at 10 s temperature of the coolant in the requisite assembly is abnormally shoot up

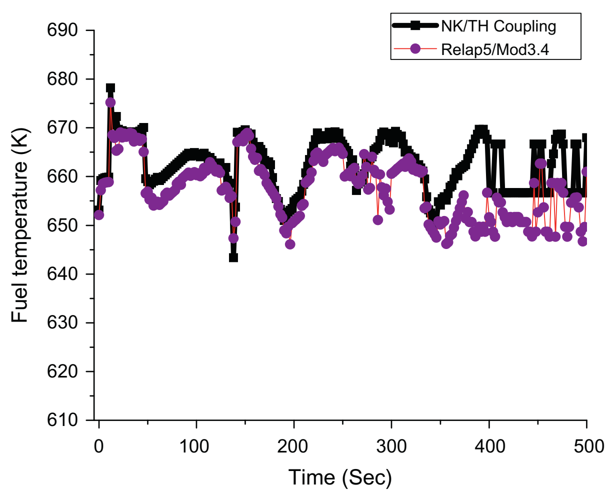

Fig. 10. Fuel temperature trends in NK/TH coupling and RELAP5 code.

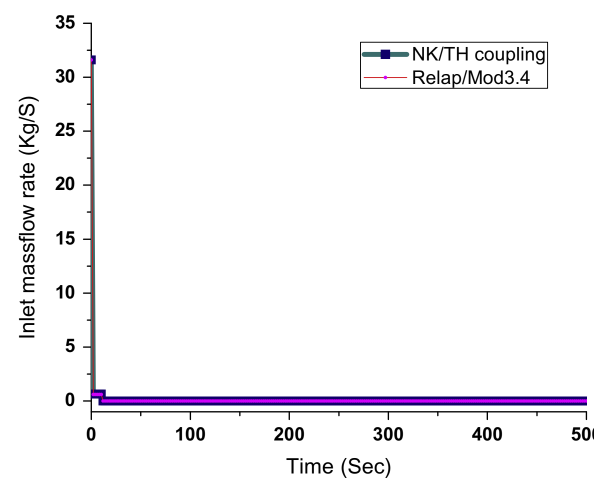

Fig. 12. Inlet mass flow rate trends in NK/TH coupling and RELAP5 code.

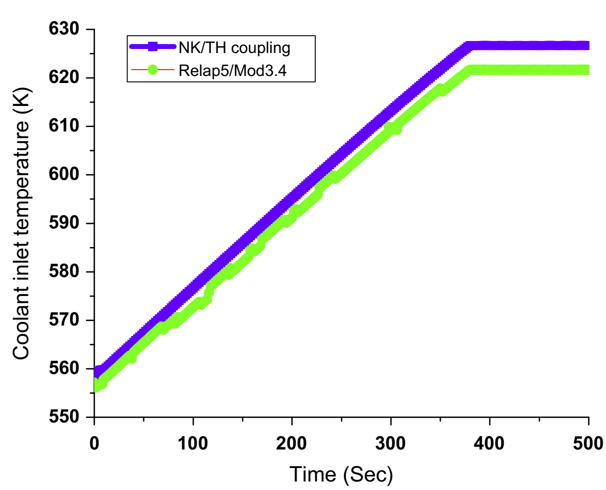

Fig. 11. Coolant inlet temperature trends in NK/TH coupling and RELAP5 code.

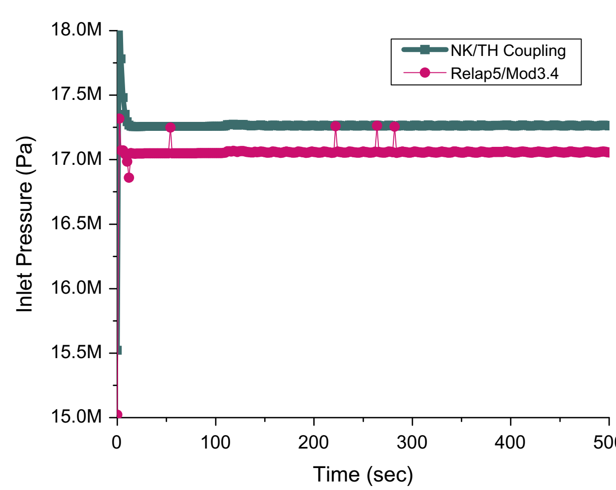

Fig. 13. Inlet pressure trends in NK/TH coupling and RELAP5 code.

to about 630 K. It can be noticed that we got a linear graph up to 400 s and then it has uniform trends. This happens due to the mass flow rate decrease in the considered assembly which then increases the coolant temperature. However, it can be seen that the after 400 s the coolant temperature is not increased rather it remains constant which leads to the conclusion that after 400 s the reactor central assembly will have same highest temperature which will definitely affect the neighboring assemblies.

In Fig. 12, it can be observed that the coolant inlet mass flow rate at the beginning of the accident suddenly goes down and reaches to a zero mass flow rate condition up to the end of an accident. The flow blockage accident can be partial as well as total blockage accident but in our case, it can be seen that there is no coolant flow across the affected assembly which then raises the temperature of fuel, coolant and clad.

In Fig. 13 it can be seen that the inlet pressure across the channel has initial values of 15.5 MPa but with the start of the accident the pressure suddenly rises and then goes down to rather intermediate values which is 17.2 MPa. After that, the pressure has the identical values as shown in Fig. 13, and this happens as the mass flow rate has negligible effects in this criteria.

As it shown in Fig. 14, coolant temperature has reached about 630 K so as with the outlet temperature which also has the same trends after the accident happened.

As shown in Fig. 12 the inlet mass flow rate trends are shown and it can be remarked that the mass flow rate goes down to zero as the accident proceeds with the time. In Fig. 15 it is concluded that the outlet mass flow rate has the flow instability trends after about 100 s. This will happen as there is no coolant flow across the first channel so there will be some minor mass flow across the neighboring assemblies.

As shown in Fig. 13, inlet pressure of the same channel has attained the same constant values with the passage of time and same is the trends in Fig. 16 due to the very less pressure difference across the assembly.

Since the flow blockage in the assembly has happened, so the reactivity will definitely have higher impact as shown in Fig. 17. At the start of the accident the negative reactivity will enlarge suddenly followed by the Doppler's effect and then have averaged trends with the passage of time due to the compensation from the other fuel assemblies.

5.1.2. Simulation of the coolant blockage accident in RELAP5

For the simulation in RELAP5 code, we employed the same nodalization technique that was used for the NK/TH coupling process. Here, we connect the time dependent valve 200TDV and

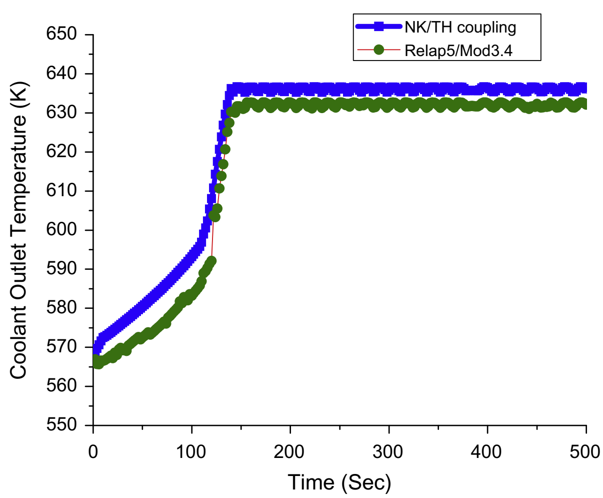

Fig. 14. Outlet coolant temperature trends in NK/TH coupling and RELAP5 code.

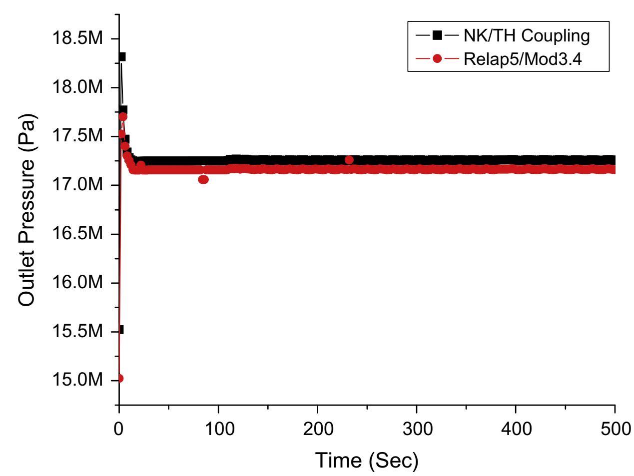

Fig. 16. Outlet pressure trends in NK/TH coupling and RELAP5 code.

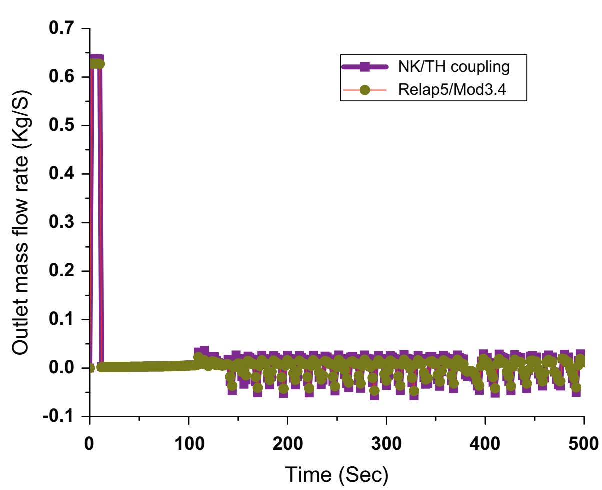

Fig. 15. Outlet mass flow rate trends in NK/TH coupling and RELAP5 code.

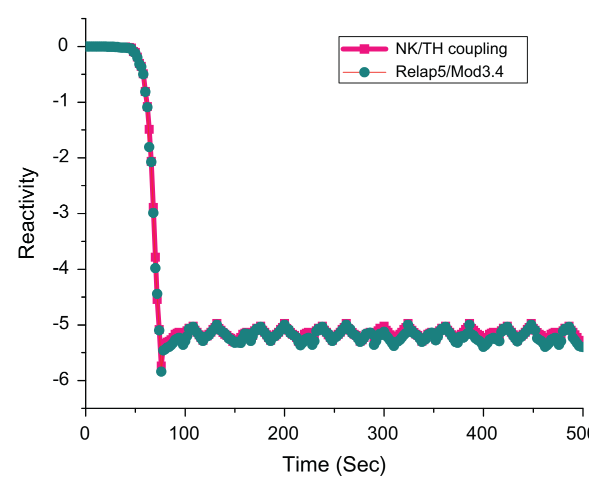

Fig. 17. Reactivity trends in NK/TH coupling and RELAP5 code.

202TDV via trip 530 as shown in Fig. 8. These valves denote the flow blockage or loss of flow accident scenario. We have checked the same parameters as were discussed in NK/TH coupling codes and find out that the results obtained from the RELAP5 code are nearly the same as presented in Figs. 9–17.

6. Conclusion

In nuclear reactor analysis, due to confined facilities for experiments which are not possible therefore comparison and validation studies have to be performed for the accuracy of the simulation results by comparing nuclear code techniques. In this paper, we analyze the safety of nuclear reactor core by the comparative approach of NK/TH coupling code technique and thermal hydraulic system code. The coupling analysis was done by using the neutron kinetic and thermal hydraulic codes which gives the power distribution of the reactor core. After we accomplished the radial and axial power distribution of the reactor core, safety studies have been performed by considering the accident scenarios. Loss of flow or coolant blockage accident was considered. The accident was simulated by using NK/TH coupling code techniques and then verified by using the thermal hydraulic system code RELAP5. The results procured from both techniques are nearly same. The comparison studies demonstrate that the coupling code technique accedes well with any other thermal hydraulic system code thus confirming the accuracy of the simulation.

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