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Effect of Temperature Difference and Control Rod Movement on Neutron Flux Distribution in Nigeria Research Reactor-1 Core

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Abstract

We described in this our work the effect of coolant temperature difference and control rod movement on neutron flux distribution in Nigerian research reactor-1(NIRR-1) core, which is miniature neutron source reactor (MNSR) in order to verify the theoretical predictions of reactor power. The results shows that the reactor neutron flux strong dependence on coolant temperature and control rod movement. These results are in perfect agreement with the constitutive model of the MNSR. A linear dependency predicted power as a function of flux parameter was observed. The rise in control rod position with time was found to compensate for neutron lost due to burn-up and thereby control the reactor flux at the preset flux values.

Keywords: Neutron flux; Temperature; Control rod; Reactor core; MNSR; NIRR-1

Introduction

In a nuclear reactor the neutron in the core of the reactor determines the dynamic behavior of the reactor. In order for the power generated by fission reactions to be maintained at a constant level the fission rate must remain steady over the course of time. In water cooled reactor like Nigeria Research Reactor-1 (NIRR-1), which is a Miniature Neutron Source Reactor (MNSR) the predominantly reactivity changes are brought about by change in the moderator (coolant temperature) and power which result in decrease in reactivity. This property is called Temperature Coefficient of Reactivity [1]. The neutron flux distribution in the core fission reactor determines the power and dynamic behavior of the reactor. The thermal neutron flux reached its peak value at the center of reactor and falls exponentially at the extreme ends of the reactor core which result to the very few thermal neutrons production in the area. Hence the flux distribution is strongest at the middle of the reactor's core [1]. However, the average flux of a reactor is a variable parameter that depends on the reactor's moderator and coolant temperature. These parameters are to be monitored for every reactor from time to time to establish the stability of the reactor's flux. It has been shown that reactor flux stability is a requirement for Neutron Activation Analysis (a major utilization of the reactor) and precious stone irradiation [2-7]. The description of Nigeria Research Reactor-1 (NIRR-1), a tank-in-pool reactor with under-moderation achieved by using a fuel material of highly enriched uranium and light water as moderator and coolant was given in our publications [3,6,8]. A major safety concern in operating a nuclear reactor is to prevent the fuel temperatures from reaching levels where they can lead to the release of fission products into the reactor vessel [8]. According to the [1] core region of NIRR-1 is located close to the bottom of reactor vessel under 4.7m of water. The 1.5m3 of water contained in the reactor vessel acts as moderator radiation shield and primary cooling medium. Reactor core cooling is achieved when water is drawn into the core through the core inlet orifice by natural convection. The water flows through the channels within the fuel elements to the top of the reactor core and finally exits through the core outlet orifice. The water heated by the fuel elements finds its way to the top of the reactor vessel, while the reactor core draws in colder water from the vessel through its inlet orifice. The heat generated by the fuel elements gives rise to the increase in reactor water temperature. This heat is extracted through the wall of the of the reactor vessel, into the 30m3 volume of water contained in the reactor pool (SAR, 2005) A temperature coefficient of reactivity is thus a desirable feature since it is an important factor in the reactor stability and operational safety. The single central control rods which perform both regulatory and shim functions in the design of NIRR-1 constitute a single point failure. The rod which is a shim rod and expected to compensate for small reactivity transient caused by change in load demand, core temperature and power level maneuvering, could not stand severe conditions [9]. The control rod cannot shutdown the Reactor when it stuck in full "out" position. Since the reactivity of the reactor is linearly related with flux and power distribution, the position of the control rod in the core at any point in time need to be monitored to establish the density or relative strength of the reactor flux [1]. Since the reactivity of the reactor is linearly related to its flux distribution determination of the control rod behavior will give information on the reactor reactivity and flux distribution. These design consideration were used in order to assess the effect of coolant temperature difference and control rod movement on neutron flux distribution on NIRR-1 core.

Theoretical Considerations

The thermal neutron flux is the quantity determine by the fission rate (1/cm^2 s) and and thus the power generated in the fuel. Due to the fact that the different at each point of the reactor, its distribution or form is of utmost importance, since it will determine the distribution of power generated in the core. The equation that determined the relation amongst core inlet temperature, coolant temperature and power level as obtained from simulation experiment on MNSR is expressed in the [10].

$$\Delta T=(5.725+147.6H^{-2.64})\, T_i^{-0.35}\, P^{(0.59+0.0019Ti)} \quad (1)$$

Where ΔT =temperature different between the inlet and outlet orifice

H= Hieght of the inlet orifice (mm)

T_i = inlet temperature oC

The designed of the inlet orifice of NIRR-1 was made to be 6mm for safety and technical reason [11-13]. Therefore subtituting the value of H into equation reduces the equation to:

$$\Delta T=7.04T_i^{-0.35}\, P^{(0.59+0.0019T_i)} \quad (2)$$

$$Thus\, P=Exp[Ln\Delta T/(7.04T_i^{-0.35})((0.59+0.0019T_i))-1] \quad (3)$$

Where ΔT=coolanttemperaturegivenby(T_0-T_i)

T_0 =outlet temperature in oC

P =predicted power kW

For a fixed height of inlet orifice it is expected from equation 3 that the reactor power varies linearly with temperature. Apart from the above method of determining the power of the reactor via thermal hydraulic parameters, (neutron flux values) could as well be exploited to predict the flux distribution of the reactor, as long as nuclear parameter are accurately known [1] the equation that relate these two parameter obeys the following relation:

$$P = 3x10^{-10} \Sigma_f V_f \phi \quad (4)$$

Where: ϕ= average thermal neutron flux in the inner irradiation channel (cm^{-2} s^{-1})

V_f = volume of the core = πr^{2}h (cm^3)

Core height (h) = 23cm

Core radius (r) = 11.5cm

Σ_f = macroscopic fission cross-section of the core fuel = 1.013x10^{-2} cm^{-1}

P= Power of Reactor in kW

A core dimension of 23cm square cylinder and highly enriched uranium was used as fuel for the Nigeria MNSR. The above parameter make it possible to reduce the equation (4) to only flux dependent parameter as shown in equation (5)

$$P = 3.0x10^{-8}\, \phi \quad (5)$$

Equation (5) reveals a linear relationship between the reactor power and its neutron flux.

Experiment

Table 1: Coolant temperature difference method.

Time (Hrs)	Inlet Temperature (OC)	Outlet Temperature (OC)	Coolant Temperature (OC)	Predicted Power (kW)	Predicted Flux (×10^{11n} Cm^{-2}s^{-1})
9:50	26.3	38.4	12.1	13.94	4.65
10:10	27.9	40.4	12.5	14.95	4.98
10:30	29.3	41.7	12.4	14.99	5
10:50	30.1	42.3	12.2	14.74	4.91
11:10	30.2	42.7	12.5	15.32	5.11
11:30	31.6	44	12.4	15.33	5.11
11:50	31.9	43.9	12	14.62	4.87

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12:10	32.1	44.2	12.1	14.83	4.94
12:30	32.2	44.1	11.9	14.47	4.82
12:50	31.5	44.3	12.8	16.09	5.36
13:10	32.6	44.4	11.8	14.34	4.78
13:30	32.7	45.2	12.5	15.67	5.22
13:50	32.2	44.1	11.9	14.47	4.82
14:10	32.6	44.7	12.1	14.9	4.97
14:30	32.6	44.9	12.3	14.28	5.09
14:50	33	44.9	11.9	14.57	4.86
15:10	33.3	45	11.7	14.24	4.75
15:30	33.1	45.2	12.1	14.96	4.99
15:50	33.3	45.1	11.8	14.42	4.81
Average	31.5	43.7	12.2	14.85	4.95

Table 2: Coolant temperature difference method.

Time (Hrs)	Inlet Temperature (OC)	Outlet Temperature (OC)	Coolant Temperature (OC)	Predicted Power (kW)	Predicted Flux ($\times10^{\wedge 11n}$ $Cm^{-2}s^{-1}$)
10:15	25	37.6	11.6	12.81	4.27
10:40	27.6	39.9	12.3	14.52	4.84
11:00	28.8	40.9	12.1	14.35	4.79
11:20	29.8	41.8	12	14.33	4.78
11:40	31.1	42.6	12.5	15.45	5.1
12:00	30	42.9	12.9	16.05	5.35
12:20	31.4	43.3	11.9	14.37	4.79
12:40	31.4	43.4	12	14.55	4.85
13:00	31.3	43.9	12.6	15.67	4.22
13:20	31.1	43.6	12.1	14.7	4.9
13:40	31.3	43.6	12.1	14.73	4.91
14:00	32.4	43.4	12.2	15.06	5.02
14:20	33.7	44.6	11.7	14.28	4.72
14:40	32.2	44.4	12.3	15.23	5.08
15:00	32.5	44.5	11.7	14.14	4.71
15:20	32.6	44.4	11.6	13.97	4.66
15:40	33	44.8	11.8	14.39	4.8
16:00	32.8	45.5	12.2	15.11	5.04
16:20	32.8	44.7	11.9	14.54	4.85
Average	32.7	43.1	12.1	14.91	4.88

Table 3: Control Movement method.

Time (Hrs)	Rod Position (Mm)	Predicted Power (kW)	Predicted Flux ($\times10^{\wedge 11n}$ $Cm^{-2}s^{-1}$)
9:50	122	15.18	5.06
10:10	125	15.18	5.06
10:30	127	15.3	5.1
10:50	132	15.41	5.14
11:10	134	15.24	5.08
11:30	135	15.27	5.09
11:50	137	15.21	5.07
12:10	139	15.33	5.11

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12:30	141	15.42	5.14
12:50	141	15.36	5.12
13:10	142	15.21	5.07
13:30	142	15.27	5.09
13:50	143	15.3	5.1
14:10	143.5	15.39	5.13
14:30	144	15.24	5.08
14:50	145	15.3	5.1
15:10	147	15.24	5.08
15:30	149	15.39	5.13
15:50	151	15.27	5.09
Average	138.9	15.29	5.1

Table 4: Control Movement method.

Time (Hrs)	Rod Position (Mm)	Predicted Power (kW)	Predicted Flux ($\times10^{11n}$ Cm-2s^{-1})
10:15	119	15.12	5.04
10:40	125	15.18	5.06
11:00	127	15.27	5.09
11:20	130	15.3	5.1
11:40	132	15.27	5.09
12:00	134	15.39	5.13
12:20	136	15.18	5.06
12:40	137	15.33	5.11
13:00	138	15.24	5.08
13:20	140	15.36	5.12
13:40	141	15.24	5.08
14:00	141	15.3	5.1
14:20	142	15.18	5.06
14:40	144	15.21	5.07
15:00	143	15.24	5.08
15:20	143	15.33	5.11
15:40	144	15.33	5.11
16:00	145	15.24	5.08
16:20	149	16.92	5.64
Average	137.3	15.35	5.11

Investigation on the effect of temperature and control rod movement on neutron flux distribution on NIRR-1 core was used as part of the ongoing research in the NIRR-1 laboratory to monitor the stability of the reactor flux required for sample activation analysis. The reactor was operated at automatic mode, in order to operate the reactor in the automatic mode, the power and the control rod limiting position of the NIRR-1 were preset to its half-power value of 15kW (flux of 5×10^{11n} cm^{-2} s^{-1}) and 220mm respectivily. This makes the reactor to operate at a power of half of its expected installed capacity. Readings were taken after every twenty minute for the two experiment for six hours and were tabulated in (Table 1-4).

Results and Discussion

The measurements were carried out at a preset flux value of 5×10^{11n} cm^{-2} s^{-1} for the whole duration of our experimental work. The result obtained shows that there is a fairly constant reactor power (15kW) and flux distribution with coolant temperature. The temperature difference recorded during the operation has also been steady at approximately 12.1 oC for the whole period of operation (Table 5). As can be seen in Tables 1-5 there is steady rise in the inlet and outlet temperature with time. This is due to the compact nature of the core, which was designed to cause insufficient thermal circulation of coolant in the core which implies that no any significant effect cause

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by heat in the flux of NIRR-1 (Figure 1 & 2). The readings in all the Tables 1-4 were collected for 6hr at a different date and time. It was observed that the averages of inlet temperatures as well as the outlet temperature were all above the room temperature which has approximate values of 32.7 oC and 43.1 oC respectively. This is in agreement with the fact that the reactor was operated not quite long to obtain the readings in (Tables 1-4) and coolant not completely cooled during the few days period of the shutdown. These values gave an average temperature difference of 12.1 oC. These temperature values did not give rise to a sharp increase in predicted flux but rather fluctuate due to the inherent features of the reactor and according to the rules governing insertion of excess reactivity into the reactor by the withdrawal of the control rod during the startup [7]. As the inlet and outlet temperature values increase, the effect of coolant temperature and control rod withdrawal on the predicted flux which shows an oscillatory behavior have an approximate values of4.92$\times10^{\wedge 11n}$ cm^{-2} s^{-1} and 5.11$\times10^{\wedge 11n}$ cm^{-2} s^{-1}respectively [14,15]. This indicated that there was no rise in the different between average values of the predicted flux and preset values. The Figure 3 show that the reactor was not completely cooled during the period of non-operation and that have no significant effect on average predicted flux of the operation which also shows the oscillation of the predicted flux around 5.00$\times10^{11n}$ cm^{-2} s^{-1} at shutdown. Furthermore the temperature difference recorded and control rod position for the two different experiment were found to be fluctuating in both cases which was constant with time due to the time delay nature shown by the flux as in Figure 4. However, the control rod position rises gradually with time to compensate for high negative temperature coefficient of reactivity in order to keep the reactor at preset power level (Table 5). This finding satisfies one of the safety requirement of the reactor which does not permit unnecessary power excursion and occurance of boiling. (Tables 1-4) reveals that NIRR-1 could be operated at half power (flux of 5$\times10^{\wedge 11n}$ cm^{-2} s^{-1} for about seven to eight continuous hours before the control rod attains it maximum value of 232mm which by implication will trigger the reactor to automatically shut down [1]. This, in combination with the limited MNSR excess reactivity that is less than 0.5$ is another safety feature guaranteed by in-build negative temperature coefficient of reactivity, [7] just in case the reactor is left unattended to for so long [16,17].

Table 5: Average values for temperature and control rod measurement.

Preset Values		Temperature Difference Method					
Power (kW)	Flux ($\times10^{\wedge 11n}$ cm^{-2}s^{-1})	Power (KW)			Flux($\times10^{\wedge 11n}$ cm^{-2}s^{-1})		
		Exp1	Exp2	Average	Exp1	Exp2	Average
15	5.00*1011	14.85	14.63	14.74	4.88	4.95	4.92
average				14.74			4.92
Control Rod Method							
Power (kW)					Flux($\times10^{\wedge 11n}$ cm^{-2}s^{-1})		
Exp1			Exp2	Average	Exp1	Exp2	Average
			15.29	15.35	5.1	5.11	5.11
average				15.32			5.11
		Deviation from preset Values		% Deviation from preset			
		Power	Flux	Power	flux		
		Exp1	Exp2	Exp1	Exp2		
		-0.03	-0.01		0.20%	0.20%	

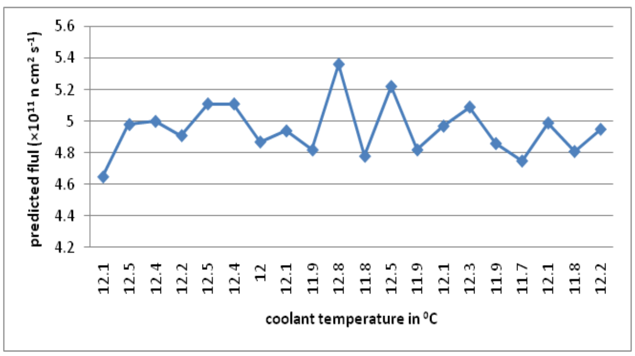

Figure 1: Variation of predicted flux with coolant temperature.

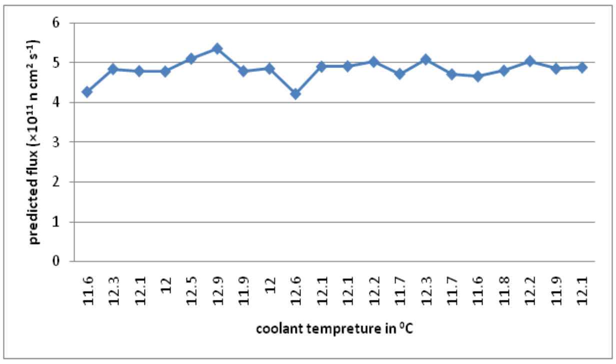

Figure 2: Variation of predicted flux with coolant temperature.

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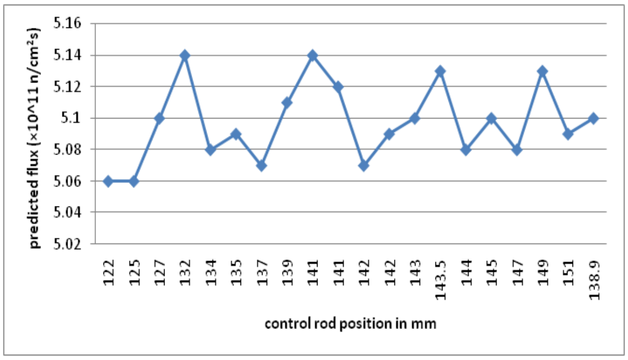

Figure 3: Variation of predicted flux with control rod position.

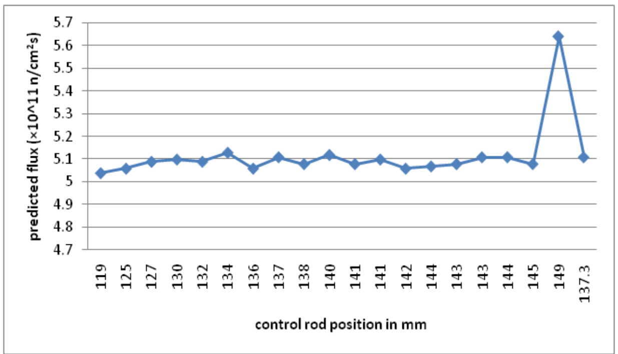

Figure 4: Variation of predicted flux with control rod position.

Conclusion

The temperature differences recorded for all the experiment conducted in this work were found to be oscillating at a constant range with time. The average predicted flux obtained for temperature and control rod methods were approximately $4.92\times10^{\wedge 11n}$ $cm^{\wedge -2}$ $s^{\wedge -1}$ and $5.11\times10^{\wedge 11n}$ $cm^{\wedge -2}$ $s^{\wedge -1}$ respectively. This shows that the flux in the core of NIRR-1 is stable. The control rod position shows a gradual rise with time. This is an in-build future to compensate for high negative temperature coefficient of reactivity in order to keep the reactor at the preset power level [7]. This result is in agreement with the safety requirement of the reactor, which does not permit power excursion and occurrence of boiling. The result obtained here also shows that mathematical relationship based on thermal hydraulics data and neurotics parameters could be used to predict the reactor's operating power level can be estimated from the preset thermal neutron flux values and vice versa.

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