Jewel Journal of Scientific Research (JJSR) 10(2): 356–363, 2025
©Federal University of Kashere-Nigeria (Print ISSN: 2384 – 6267, Online ISSN 2756-651X)
https://journals.fukashere.edu.ng/index.php/jjsr

Effect of Poison and Thermal Hydraulic on Miniature Neutron Source Reactor (MNSR) Parameters

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Abstract
Over the years of operation of the Nigeria Miniature Neutron Source Reactor's after installation in 2004, numerous installations and calculations were made for the safety of Nigeria Research Reactor-1 in accordance with project supply agreements and IAEA member states. By using the lattice code WIMS and the core analysis code CITATION to automatically perform each of the following calculations on MNSRs: core excess reactivity, verification of certain safety criteria, and calculation of various temperature coefficients of reactivity, we demonstrate in this work the impact of removing cadmium poison on neutron flux and related parameters. Results obtained before, after and after removal of cadmium poison are: excess core reactivity (3.72, 2.96 and 2.92) mk, fuel temperature coefficient reactivity (−0.0018, −0.0054 and −0.0060) mk/˚C, power coefficient of reactivity (−0.2527, −0.1260 and 0.0575) mk/kW and predicted power of (15.23, 14.65 and 14.99) kW with coolant temperature (12.43, 12.10 and 12.50) ˚C respectively. The result will not only boost the sample handling capabilities of NIRR-1 but will also provide useful data to the MNSR community for upgrading their reactors and specifically ageing management.

Keywords: Fuel temperature coefficient reactivity; Power coefficient of reactivity; WIMS and citation

Received: 10th Sept, 2025 *Accepted:* 28th Nov, 2025 *Published Online:* 27th Dec 2025

Introduction

The thermal neutron flux is highest in the reactor core, but decreases at the extreme extremities because very few thermal neutrons are created in this area. As a result, the flux distribution is most concentrated in the reactor's core Ahmed *et al*., 2008 and Abrefah *et al*., 2010. However, a reactor's average flux is a changeable parameter that is determined by the moderator and coolant temperature. These parameters must be evaluated on a regular basis for each reactor to ensure that the reactor's flux remains stable.

Over the years, determining the dynamic behavior of reactors such as flux parameters, excess reactivity, fuel temperature coefficient of reactivity (FTCR), and power coefficient of reactivity (PCR) in water-moderated reactors such as the Nigeria Research Reactor - 1 (NIRR-1) has been an important operational parameter associated with safety consideration. The (FTCR) is defined as the change in reactivity per degree change in the core-averaged moderator temperature, and the (PCR) is defined as the change in reactivity per degree change in the reactor's core power, i.e. the power coefficient of reactivity (PCR) is the total reactivity change between the two states rather than the reactivity change per degree Celsius (°C).

The Nigeria Research Reactor-1 (NIRR-1) is a Miniature Neutron Source Reactor (MNSR) located at Ahmadu Bello University's Center for Energy Research and Training in Zaria. The reactor features a tank-in-pool structural arrangement and uses 90.2% highly enriched uranium (HEU) as fuel clad with aluminum, light water as a moderator and coolant, and a metal beryllium encircling the reactor's core as a reflector Ahmed *et al*., (2008), Jonah *et al*., (2011) and Khamis and Alhalabi (2007). The thermal power of NIRR-1 is 31 kW, with a thermal neutron flux of 1×10^{12} $cm^{-2}.s^{-1}$. The reactor core is a square cylinder of 230 × 230 mm, with 347 fuel pins and three Al dummies in the fuel lattice. The fuel element has an active length of 230 mm and 9 mm Al-alloy plugs on both ends Jonah *et al*., (2007). The reactor is refueled by replacing the entire core with a new one, which has a lifespan of more than ten years Gao *et al*., (1992). In NIRR-1, only one control rod is accessible, which also serves as a regulation, shim, and safety rod. The reactor achieves operating power by gradually removing the control rod, maintains operating power by adjusting the control rod, and shuts down by returning the rod to the core Jonah *et al*., (2011) and Ahmed *et al*., (2011).

During the NIRR-1's startup, criticality experiments were conducted and precise adjustments were made to obtain the excess reactivity of 3.77×10^3 Δk/k. To meet these requirements, the first part of the start-up was carried out at the zero-power facility at the China Institute of Atomic Energy in Beijing, and the second stage saw the reactor being started on-site at the Centre for Energy Research and Training in Zaria. The following tests Balogun *et al*., 2004 included both off-site zero power and on-site reactor starting. Since NIRR-1's inauguration in 2004, many studies have been done to assess its safety parameters. The change in the thermal neutron spectrum with temperature will alter the balance between the fission and absorption rates in the core, because the fission and absorption cross-sections are both functions of the neutron energy. The fuel temperature coefficient arises principally from two factors: one is the Doppler effect and the other the effect due to the fact that the neutron spectrum is slightly hardened by the increase in fuel temperature. This might generate coolant temperature increase due to excessive power, low flow and low pressure.

The condition of coolant temperature increase will gradually be depleted at some channels which create some void in some channels, producing less moderation for interaction of fuel with high energy neutron, creating large positive reactivity. However, the safety feature of NIRR-1, excess negative reactivity will quickly reduce the power to balance increase in reactivity. It is the aim of this research work to check the effect of cadmium poison and thermal hydraulic on neutron parameter on MNSR.

Theoretical Considerations

The equation that gives the relationship between core temperature difference, inlet temperature and power level as obtained from simulation experiment on MNSR is given as Ahmed *et al*., (2008) and Hainoun and Alisa, (2005):

$$\Delta T \ = T\ (5.725 + 147.6H^{-2.64}\ )T_i^{-0.35}\ p(0.59 + 0.0019T_i\ ) \tag{1}$$

Where, ΔT = temperature different between the inlet and outlet orifice (˚C)

H = Hieght of the inlet orifice (mm)

T_i = inlet temperature (˚C)

The inlet orifice of the NIRR-1 "H" is 6 mm Ahmed *et al*., (2008), Khamis and Alhalabi (2007), Yang, (1992). Therefore, putting the value of H into equation (1) the predicted power "P" and thermal limitation of the reactor was determined.

According to the simplified dynamics model, the reactivity ρ(t) that drives the transients is the net effect of contributions arising from several mechanisms. These are represented as Alhassan *et al*., (2011).

$$\rho(t) = \rho_i\ (t) + \rho_{fb}(t) + \rho_c(t) + \rho_{sd}\ (t) \tag{2}$$

where: $\rho_i(t)$ is the reactivity caused by the initiating event, $\rho_{fb}(t)$ the reactivity from thermal e hydraulics feedback, $\rho_c(t)$ the reactivity from the reactor power control systems, and $\rho_{sd}(t)$ the shutdown or trip reactivity. Based on the simplified dynamics model, feedback reactivity changes, $\rho_{fb}(t)$, may be represented in terms of the T_{fe}; T_c and T_i that are the core-averaged fuel element, coolant, and inlet temperature coefficients of (feedback) reactivity, respectively Alhassan, (2007).

$$d\rho_{fb}(t) = \frac{1}{K}\frac{\partial K}{\partial T_{fb}}d\overline{T_{fb}} + \frac{1}{K}\frac{\partial K}{\partial T_c}d\overline{T_c} + \frac{1}{K}\frac{\partial K}{\partial T_i}d\overline{T_l} \quad (3)$$

For very short time that the heat cannot transferred to the coolant, the reactor fuel element may be considered to behave adiabatically.

Fuel Temperature coefficient of reactivity (FTCR)

In a homogeneous reactor, the temperature influence on reactivity can be stated as a simple temperature coefficient, α_{fuel}, which represents the change in reactivity per degree change in temperature. The fuel temperature coefficient of reactivity (FTCR) is calculated by dividing the change in reactivity ($\Delta\rho$) by the change in fuel temperature (ΔT_{fuel}) Lamarsh and Baratta, (1982).

$$\alpha_{fuel} = \frac{\Delta\rho}{\Delta T} \quad (4)$$

The reactivity change $\Delta\rho$ is determined as

$$\Delta\rho = \frac{K_2 - K_1}{K_2 K_1} \quad (5)$$

Where: k1 and k2 are respectively the effective multiplication factors for reactor states 1 and 2. The multiplication factors are determined with the help of static criticality calculations, performed with a two-dimensional, two-energy group model Alhassan *et al*. (2011).

Power reactivity coefficient

When a reactor operates at more than a few percent of its rated power, it is no longer realistic to assume homogeneous temperature throughout the core. The coolant temperature rises as it goes through the core, and the fuel element temperature must be significantly higher than that of the coolant to promote fuel-to-coolant heat transfer Alhassan *et al*., (2011). The incremental change in reactivity with power due to temperature differences. To understand how the operating power level affects the core's reactivity, imagine that the mean temperature of the coolant and its rate of flow remain constant throughout all power levels. An increase in power level must then cause an increase in the fuel temperature, where it is generated, into the coolant that carries it away. Under these conditions it is obvious that an increase in power level will cause a negative change in the reactivity. The power coefficient of reactivity is defined as Alhassan, (2009) and Erfani *et al*., (2012)

$$\alpha_p = \frac{\Delta\rho}{\Delta P} \quad (15)$$

Where, ΔP is the change in power and $\Delta\rho$ is change in the reactivity of the reactor.

Experimental Methodology

The Nigerian Research Reactor-1 was operated at a fixed flux value of 5×10^{11} n cm^{-2} s^{-1} with a control rod position of 220 mm. This causes the reactor to run at half of its estimated installed capacity. Readings were taken every twenty minutes for six hours and are presented in Tables 1, 2, and 3.

The reactor's coolant input and outlet temperatures were recorded every 20 minutes for six hours at a neutron flux of 5 $\times10^{11}$ n cm^2 s^{-1}. The temperature difference between the two was then calculated.

To verify whether at higher fuel temperatures, neutrons with speed slightly “off” the resonance energy can still be absorbed in the resonance.

To verify if the effect of resonance is broadened at higher temperature (Doppler broadening).

Even though the resonance peak is at the same time lowered somewhat, the overall result is that there is more absorption in the resonance at higher fuel temperature.

Results and Discussions

In this experiment, the reactor was shut down for one week, so the reactor was xenon poisoning free with low background signal Do Prado Souza *et al*., (2006). When the temperature was found to be steady, the reactor was started-up and went to a preset power of 15 kW, following normal start-up procedures. The reactor stayed at a steady power level of 15 kW for the period of operation. The average value for the predicted power of the reactor before, after and after removal of cadmium poison at half flux were found to be (15.85 ± 0.85) kW, (14.64 ± 0.36) kW and (14.99 ± 0.01) kW with average values of coolant temperature of 12.43 ˚C, 12.10 ˚C and 12.5 ˚C respectively.

Cadmium line poison was used at the Nigeria research reactor-1 to boost reactivity, which was found to be declining after six years of operation to a value of 2.64 mk. This demands the installation of a cadmium line to boost the reactor's excess reactivity. Because of the high demand for epithermal and thermal neutrons by clients of the Nigerian Research Reactor-1 (NIRR-1), a Miniature Neutron Source Reactor, the control rod was removed from the core and the reactor was made critical using cadmium shim capsules. Keeping the reactor in auto mode, the system reactivity was determined when half of the control rod was removed. This test was done before, after and after removal of cadmium poison. After successful installation cadmium line of the reactor, tests were performed to check the safe operation of the reactor. The calculated value of excess reactivity for the cold clean core was found to be within the measured value with an average value of 3.72 mk (Table 4). The value of excess reactivity after installation of cadmium poison was found to be 2.96 mk. After two years of operation, it was also observed that the core excess reactivity has decrease to value of 2.2 mk. After carefully considering the safety implication, the reactor manager recommended the withdrawal of the cadmium poison in an inner irradiation channel of the reactor. That was done and the cadmium poison positioned in the reactor pool to cool down. After successful exercise, the core excess reactivity increased 2.92 mk (Table 4). The reactivity change was derived from the position of the Shim rod. All Table 1, 2 and 3 present the results obtained. The results are presented in Table 1, 2 and 3. The value of the reactivity introduced, and the power level, provided the date for determining the average power coefficient of reactivity in the power interval involved equation 15 Do Prado Souza *et al*., (2006).

Table 1: NIRR-1 clean cold-core excess reactivity in mK, power and fuel coefficient of reactivity at a neutron flux of 5×10^{12} ncm^{-2} s^{-1}.

Time (hrs)	**Predicted flux ($\times10^{11}$ cm-2 s-1)**	**Rod position (mm)**	**Inlet temperature (˚C)**	**Outlet temperature (˚C)**	**Coolant temperature Difference (˚C)**	**Predicted power (kW)**	**ΔP (kW)**	**P (mK)**	**Δρ (mK)**	**αfuel (mK/°C)**	**α_p (mK/kW)**
10:15	5.03	146.0	27.9	40.6	12.7	15.30	0.00	3.32	0.00	0.0000	0.0000
10:40	5.10	151.0	29.4	41.9	12.5	15.39	0.09	3.50	0.18	0.0144	2.0000
11:00	5.13	153.0	30.0	42.7	12.7	15.09	−0.30	3.40	−0.10	−0.0079	0.3333
11:20	5.09	153.0	30.6	42.9	12.3	15.09	0.00	3.40	0.00	0.0000	0.0000
11:40	5.03	153.5	30.9	43.6	12.7	15.15	0.06	3.54	0.14	0.0110	2.3333
12:00	5.02	154.5	31.1	43.4	12.3	15.18	0.03	3.58	0.04	0.0033	1.3333
12:20	5.05	157.0	31.8	44.1	12.3	15.12	−0.06	3.69	0.11	0.0089	−1.8333
12:40	5.06	157.5	31.5	44.6	12.6	15.06	−0.06	3.71	0.02	0.0016	−0.3333
13:00	5.04	158.0	31.3	43.9	12.6	15.27	0.21	3.72	0.01	0.0008	0.0476
13:20	5.02	159.0	31.0	44.4	13.4	15.27	0.00	3.76	0.04	0.0030	0.0000
13:40	5.09	160.0	32.2	44.5	12.3	15.30	0.03	3.79	0.03	0.0024	1.0000
14:00	5.09	160.0	32.5	44.7	12.2	15.21	−0.09	3.79	0.00	0.0000	0.0000
14:20	5.10	161.0	32.0	45.0	13.0	15.33	0.12	3.81	0.02	0.0015	0.1667
14:40	5.07	161.5	32.7	45.1	12.4	15.33	0.00	3.84	0.03	0.0024	0.0000
15:00	5.11	162.0	32.7	45.1	12.4	15.24	−0.09	3.85	0.01	0.0008	−0.1111
15:20	5.11	162.0	32.7	44.8	12.1	15.30	0.06	3.85	0.00	0.0000	0.0000
15:40	5.08	163.0	32.3	45.1	12.8	15.48	0.18	3.89	0.04	0.0031	0.2222

16:00	5.10	163.0	33.0	45.2	12.2	15.18	−0.30	3.89	0.00	0.0000	0.0000
16:20	5.16	163.5	33.1	45.2	12.1	15.09	−0.09	3.90	0.01	0.0008	−0.1111
16:40	5.06	164.0	32.6	45.0	12.4	15.18	0.09	3.90	0.00	0.0000	0.0000
17:00	5.03	165.0	33.2	44.5	11.6	15.45	0.27	3.94	0.04	0.0034	0.1481
17:20	5.15	166.0	33.3	45.1	11.8	15.12	−0.33	3.82	−0.12	−0.0102	0.3636
average	5.08					15.23		3.72		−0.0018	0.2527

Table 2: NIRR-1 core excess reactivity in mK, power and fuel coefficient of reactivity at a neutron flux of 5×10^{12} $ncm^{-2}s^{-1}$ after installation of cadmium line.

Time (hrs)	**Predicted flux ($\times10^{11}$ n cm-2 s-1)**	**Rod position (mm)**	**Inlet temperature (˚C)**	**Outlet temperature (˚C)**	**Coolant temperature Difference (˚C)**	**Predicted power (kW)**	**ΔP (kW)**	**ρ(mK)**	**Δρ(mK)**	**αfuel(mK/°C)**	**α_p(mK/kW)**
10:15	4.27	119	25	37.6	11.6	12.81	0.00	2.16	0.00	0.000	0.000
10:40	4.84	125	27.6	39.9	12.3	14.53	1.71	2.43	0.27	0.022	0.158
11:00	4.79	127	28.8	40.9	12.1	14.36	−0.17	2.53	0.10	0.008	−0.587
11:20	4.78	130	29.8	41.8	12.0	14.33	−0.03	2.67	0.14	0.012	−4.459
11:40	5.10	132	31.1	42.6	12.5	15.45	1.13	2.74	0.07	0.006	0.062
12:00	5.35	134	30.0	42.9	12.9	16.05	0.60	2.82	0.08	0.006	0.133
12:20	4.79	136	31.4	43.3	11.9	14.37	−1.69	2.91	0.09	0.008	−0.053
12:40	4.85	137	31.4	43.4	12.0	14.55	0.19	2.96	0.05	0.004	0.269
13:00	4.22	138	31.3	43.9	12.6	15.67	1.12	3.00	0.04	0.003	0.036
13:20	4.90	140	31.1	43.6	12.1	14.70	−0.98	3.07	0.07	0.006	−0.072
13:40	4.91	141	31.3	43.6	12.1	14.73	0.03	3.13	0.06	0.005	2.166
14:00	5.02	141	32.4	43.4	12.2	15.06	0.34	3.13	0.00	0.000	0.000
14:20	4.72	142	33.7	44.6	11.7	14.28	−0.78	3.15	0.02	0.002	−0.026
14:40	5.08	144	32.2	44.4	12.3	15.23	0.94	3.24	0.09	0.007	0.095
15:00	4.71	143	32.5	44.5	11.7	14.14	−1.09	3.21	−0.03	−0.003	0.028
15:20	4.66	143	32.6	44.4	11.6	13.97	−0.17	3.21	0.00	0.000	0.000
15:40	4.80	144	33.0	44.8	11.8	14.39	0.42	3.24	0.03	0.003	0.071
16:00	5.04	145	32.8	45.5	12.2	15.11	0.73	3.26	0.02	0.002	0.027
16:20	4.85	149	32.8	44.7	11.9	14.54	−0.58	3.40	0.14	0.012	−0.242
Average	4.83					14.65		2.96		−0.005	−0.126

Table 3: NIRR-1 core excess reactivity in mK, power and fuel coefficient of reactivity at a neutron flux of 5×10^{12} ncm^{-2} s^{-1} after removal of cadmium poison.

Time (hrs)	**Predicted flux ($\times10^{11}$ n cm-2 s-1)**	**Rod position (mm)**	**Inlet temperature (˚C)**	**Outlet temperature (˚C)**	**Coolant temperature Difference (˚C)**	**Predicted power (kW)**	**ΔP (kW)**	**ρ(mK)**	**Δρ(mK)**	**α_{fuel}(mK/0C)**	**α_p(mK/kW)**
11:15	5.91	129	25.6	38.3	12.7	14.89	0.00	2.62	0.00	0.0000	0.0000
11:25	4.90	132	27.1	39.5	12.4	14.62	−0.27	2.74	0.12	0.0097	−0.4444
11:35	5.90	133	27.7	40.3	12.6	15.10	0.48	2.80	0.06	0.0048	0.1250

11:45	5.96	135	28.8	40.9	12.1	14.36	−0.74	2.86	0.06	0.0050	−0.0811
11:55	6.02	137	29.3	41.6	12. 3	14.81	0.45	2.96	0.10	0.0081	0.2222
12:05	5.97	140	29.7	41.7	12.0	14.31	−0.50	3.07	0.11	0.0092	−0.2200
12:15	6.00	141	29.7	42.8	13.1	16.39	2.08	3.13	0.06	0.0046	0.0288
12:25	6.00	143	30.2	42.8	12.6	15.51	−0.88	3.21	0.08	0.0063	−0.0909
Average	5.83				12.5	14.99		2.92		−0.0060	−0.0575

Table 4: Summary of result for the Nigeria research reactor-1 parameters before, after and after removal of cadmium poison

		Calculated values	**Measured Values**
Coolant temp.	Before	12.4 ˚C	
	After	12.1 ˚C	12.1 ˚C Ahmed *et al*., (2008)
	After Removal	12.5 ˚C	
Predicted power	Before	15.23 kW	
	After	14.65 kW	15 kW
	After Removal	14.99 kW	
Excess reactivity	Before	3.72 mk	
	After	2.96 mk	3.5 − 4.0 mk
	After Removal	2.92 mk	
Fuel temp. Coefficient. of reactivity	Before	−0.0018 mk/^{0}C	
	After	−0.0054 mk/^{0}C	Negative Balogun, (2003)
	After Removal	−0.0060 mk/^{0}C	
Power coefficient of reactivity	Before	−0.2527 mk/kW	
	After	−0.1260 mk/kW	not available yet
	After Removal	−0.0575 mk/kW	

In order to obtain the fuel power coefficient of reactivity, a correction based on the experiment described above was applied for the rise in coolant temperature during the operation. The approximate values of the power coefficient of reactivity before, after and after removal of cadmium poison were found to be (−0.2527, −0.1260 and −0.0575) mk/kW respectively. Similarly, the fuel temperature coefficient of reactivity was determined to be negative with average of (0.0018, −0.0054 and −0.0060) mk/˚C respectively Balogun, (2003). Comparing the power coefficient of reactivity, it is noted that the rise in coolant temperature has contributed only with a small fraction to the observed negative effect as shown in the table, these values also show the associated reactivity loss to achieve a given power level. The curve is almost linear and gives approximate values power coefficient stated above. Because of the prompt negative temperature coefficient, a significant amount of reactivity is needed to overcome temperature and allow the reactor to operate at higher power levels in steady state operation.

Finally, during the operation of the reactor, the inlet temperature was also recorded in all the tables which show that the values are in good agreement with the findings of Ahmed *et al*. 2008. However, as time increases, some kind of balance is achieved between heat generation in the core and heat being lost due to cooling of the core and partially accumulated in the water of the reactor vessel

Khamis and Jamal, (2006). 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, Ahmed *et al*., 2011 and Muhammad *et al*., (2001) just in case the reactor is left unattended to for so long and all parameters of the reactor are within the design limit.

Conclusion

Measurements were taken in the Nigeria research reactor-1 (NIRR-1), which has 347 fuel pins in its core. Our results demonstrate that the reactor's excess reactivity prior to the cadmium-liner installation was reported to be 2.64 mk Balogun *et al*., (2004) and Abrefah *et al*., (2010). The trials carried out here before, during, and after installation allowed us to attain an average core excess reactivity of 3.72 mk, 2.96 mk, and 2.92 mk, respectively. Because of the prompt negative temperature coefficient, the reactor requires a substantial amount of reactivity to overcome temperature and run at higher power levels. This increases NIRR-1's reactivity margin significantly without the requirement for a beryllium shim, as might have been the case. The results of neutron flux measurements taken before, during, and after removal of the installation reveal that there are no substantial fluctuations in flux, coolant temperature, or reactor power. The power coefficient of reactivity was determined before, after, and after removing the installation of cadmium poison, with average values of (-0.2527, -0.1260, and -0.0575) mk/kW accordingly, with negative fuel temperature coefficient of reactivity. As previously demonstrated, the majority of the negative reactivity shift with increasing power can be attributable to the change in fuel temperature (prompt coefficient). The delayed coefficient owing to water heating was extremely modest. Then, we found that the power coefficient of reactivity of the fuel is the primary contributor to the power coefficient of reactivity Do Prado Souza et al., (2006). It has been established that this work will greatly aid in the safe and optimal utilization of epithermal and thermal neutrons for neutron activation analysis. Finally, this study demonstrates that the thermal neutron flux and other determined parameters of the channel did not depart considerably from the preset flux which shows the stability and safety of NIRR-1 after deinstallation of cadmium poison.

Acknowledgements

The authors wish to thank the operation staff of NIRR-1 for their cooperation during the experimentation.

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