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EVALUATION OF THE IMPACT OF CADMIUM POISON AND CADMIUM LINE CHANNEL ON THE DYNAMIC BEHAVIOR OF MINIATURE NEUTRON SOURCE REACTOR (MNSR)

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ABSTRACT
Nigerian Research Reactor-1 (NIRR-1) undergo different installation such as permanent Cadmium line due to high demand for epithermal neutrons activation analysis (ENAA) irradiation by the clients and cadmium poison to regulate the reactivity of the reactor with a range value 4.0mk to 3.7mk. Safety and flux stability. Simulations were carried out via Monte Carlo Transport Code MCNP5 for these installations but very few experimental calculation were done. The results of all these research work revealed that the effect of cadmium-line on safety and flux stability is very small. But in the case of cadmium poison, the reactivity of the reactor after 10 years of operation was observe to be decreasing. The results obtained for excess core reactivity after (2.96 and 2.95) mk, and predicted power of 14.65 kW for the two experiment with coolant temperature (12.10 and 12.08) ˚C respectively. These show that the change in power of the reactor is very small. The outcome of the research will help sample handling capabilities of NIRR-1 and provide useful data to the MNSR.
Key words: Epithermal neutron, ENAA Irradiation, NIRR-1,

INTRODUCTION

The thermal neutron flux achieves its maximum value in the core of reactor but reduced at extreme ends of the reactor core, since very few thermal neutrons are produced in this area. Hence the flux distribution is strongest at the middle of the reactor's core (Ahmed *et al.*, 2008; Musa *et al.*, 2011). However, the average flux of a Reactor is a variable parameter that depends on the Reactor's moderator and coolant temperature. Epithermal neutron activation analysis (ENAA) method is a technique which is ideally suited for determination of key trace elements in geological, biological and environmental samples which cannot be accomplished by traditional use of thermal neutrons alone (Ahmed *et al.*, 2013). The ENAA technique is one of the most highly developed techniques in experimental nuclear physics (Ahmed *et al.*, 2013). The method became popular because gamma ray detection and gamma spectrum determination which forms the essential measurement at every step in nuclear spectroscopic experiments was optimized in recent years (Ahmed et al., 2010). The advantages pertaining to the potential use of epithermal neutrons to overcome background interferences from thermally activated elements has been generalized in a useful form by (Decorte et al. 1972). Over the years, determination the dynamic behaviour of reactors such as flux parameters, excess reactivity, in MNSR is an important operational parameter connected with safety considerations. During a start-up of the NIRR-1, the criticality experiments were performed and fine adjustments were made to achieve the excess reactivity of 3.77 × 103 Δk/k (cadmium poison installation). In order to satisfy these requirements, the first stage of the start-up was performed at the zero power facility at China Institute of Atomic Energy, Beijing and the second stage on-site start-up of the reactor at the Centre for Energy Research and Training, Zaria. The o••site zero power and the on-site startup of the reactor were both performed for the following tests (Balogun *et al.*, 2004). Other MNSR groups have demonstrated the dependence of neutron flux and power of MNSR on thermal hydraulic parameters where they have reported that the installations could affect parameters such reactor Power Reactivity and flux distribution (Khamis and Jamal 2006; Hainoun and Alisa 2005 and Khamis and Alhalabi 2007). These parameters are to be monitored for every Reactor from time to time to establish the stability of the reactor's flux.

Theoretical Considerations

The thermal neutron flux is the amount decide by the fision rate $(1/cm^2 s)$ and along these lines, the power generated in the fuel.

Because of the way that the distinctive at each purpose of the reactor, its distribution is of most extreme significance, since it will determine the distribution of power created in the core (Jonah *et al.*, 2006 Ahmed *et. al.*, 2011; 2008).

The equation that determine the connection among core inlet temperature, coolant temperature and power level as obtained from simulation experiment on MNSR is expressed in the form. (Anas *et al.*, 2015; Ahmed *et. al.*, 2011; 2008; Shi, 1990).

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

Where ΔT =temperature different between the inlet and outlet orifice

H= Hieght of the inlet orifice (mm)

T_i = inlet temperature °C

The designed of the inlet orifice of NIRR-1 was made to be 6mm for safety and technical reason (Yang, 1992; khamis and Jamal 2006, Khamis and Alhalabi 2007, (Ahmed *et. al.*, 2008; 2008)). Therefore subtituting the value of H into equation (1) reduces the equation to:

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

Thus$P = Exp\left[Ln\left(\frac{\Delta T}{7.04T_i^{-0.35}}\right)((0.59 + 0.0019Ti{-}1\right. \quad (3)$

Where $\Delta T = coolant temperature given by (T_0 - T_i)$

T_0 = outlet temperature in °C

P = predicted power kW

For a settled tallness of inlet orifice, it is normal from equation 3 that the reactor power varies directly with temperature.

Aside from the above strategy for determining the power of the reactor by means of thermal pressure driven parameters, (neutron flux values) could too be used to predict the flux distribution of the reactor as long as nuclear parameter are precisely known (Ahmed *et. al.*, 2008) the condition that relate these two parameter complies with the accompanying equation:

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

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

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

Core height (h) = 23cm

Core radius (r) = 11.5cm

$\sum_f$ = macroscopic fission cross-section of the core fuel = 1.013x$10^{-2}cm^{-1}$

P= Power of Reactor in kW

A core measurement of 23cm square cylinder and highly enriched uranium was utilized as fuel for the Nigeria MNSR. The above parameter make it conceivable to diminish the equation (4) to just transition subordinate parameter as appeared in equation (5)

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

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

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

$$\rho(t) = \rho_i(t) + \rho_{fb}(t) + \rho_c(t) + \rho_{sd}(t) \quad (6)$$

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.(Alhassan *et al.*, 2009).

$$\rho = \frac{k-1}{k} \quad (7)$$

The reactivity change Δρ is determined as

$$\Delta\rho = \frac{k_2 - k_1}{k_2 k_1} \quad (8)$$

where: k_1 and k_2 are respectively the e¤ective 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 CITATION CODE (Alhassan *et al.*, 2011).

Experiment

Examination on the Evaluation of the Impact of Cadmium poison and Cadmium line Channel on the dynamic behavior of Miniature Neutron Source Reactor (MNSR) was utilized as a component of the progressing research in the NIRR-1 laboratory to screen the dependability of the reactor transition required for test initiation investigation. The reactor was worked at programmed mode, so as to work the reactor in the programmed mode, the power and the control pole constraining position of the NIRR-1 were preset to its half-control estimation of 15kW (flux of 5×10^{11}n $cm^{-2}s^{-1}$) and 220 mm respectivily. This makes the reactor to work at an intensity of half of its normal power limit. (Anas *et al.*, 2015) Readings were taken after each twenty minute for the two trial for six hours and were arranged in Table (1-2).

RESULTS AND DISCUSSION

In this work, the reactor was shut down for a week, so the reactor was xenon poisoning free (Do prado *et al.*, 2006). At the point when the temperature was observed to be relentless, the reactor was started up and went to a preset power of 15 kW, following typical start-up methodology. The reactor remained at a consistent power level of 15 kW for the time of experiment.

Table 1: NIRR-1 predicted power and flux, coolant temperature difference and core excess reactivity in mK, at a neutron flux of $5{\times}10^{11}$ n $cm^{-2}s^{-1}$ 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)	ρ(mK)
10:20	4.57	119	25.0	37.6	11.6	12.81	2.16
10:40	4.79	125	27.6	39.9	12.3	14.53	2.43
11:00	4.89	127	28.8	40.9	12.1	14.36	2.53
11:20	4.68	130	29.8	41.8	12.0	14.33	2.67
11:40	5.00	132	31.1	42.6	12.5	15.45	2.74
12:00	5.45	134	30.0	42.9	12.9	16.05	2.82
12:20	4.70	136	31.4	43.3	11.9	14.37	2.91
12:40	4.95	137	31.4	43.4	12.0	14.55	2.96
13:00	4.29	138	31.3	43.9	12.6	15.67	3.00
13:20	4.90	140	31.1	43.6	12.1	14.70	3.07
13:40	4.97	141	31.3	43.6	12.1	14.73	3.13
14:00	5.02	141	32.4	43.4	12.2	15.06	3.13
14:20	4.72	142	33.7	44.6	11.7	14.28	3.15
14:40	5.04	144	32.2	44.4	12.3	15.23	3.24
15:00	4.71	143	32.5	44.5	11.7	14.14	3.21
15:20	4.66	143	32.6	44.4	11.6	13.97	3.21
15:40	4.70	144	33.0	44.8	11.8	14.39	3.24
16:00	5.04	145	32.8	45.5	12.2	15.11	3.26
16:20	4.83	149	32.8	44.7	11.9	14.54	3.40
average	4.84	137	31.1	43.1	12.1	14.65	2.96

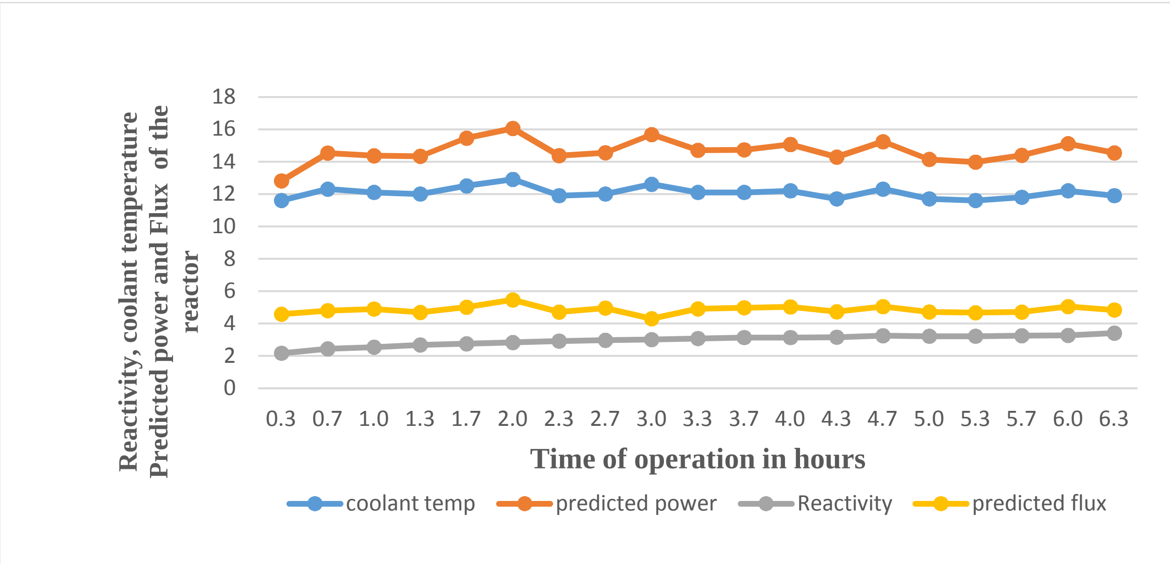

Figure 1: Variation of reactivity, coolant temperature, predicted power and flux with time of operation

Table 2: NIRR-1 predicted power and flux, coolant temperature difference and core excess reactivity in mK, at a neutron flux of 5×10[11] $\mathrm{n}\ cm^{-2}s^{-1}$ after installation of cadmium line.

Time (hrs)	Predicted flux (×10[11] $\mathrm{n}\ cm^{-2}s^{-1}$)	Rod position (mm)	Inlet temperature (˚C)	Outlet temperature (˚C)	Coolant temperature Difference (˚C)	Predicted power (kW)	ρ(mK)
09:20	4.57	119	25.0	37.6	11.6	12.81	2.16
09:40	4.79	125	27.6	39.9	12.3	14.53	2.43
10:00	4.89	127	28.8	40.9	12.1	14.36	2.53
10:20	4.68	130	29.8	41.8	12.0	14.33	2.67
10:40	5.00	132	31.1	42.6	12.5	15.45	2.74
11:00	5.45	134	30.0	42.9	12.9	16.05	2.82
11:20	4.70	136	31.4	43.3	11.9	14.37	2.91
11:40	4.95	137	31.4	43.4	12.0	14.55	2.96
12:00	4.29	138	31.3	43.9	12.6	15.67	3.00
12:20	4.90	140	31.1	43.6	12.1	14.70	2.89
12:40	4.93	141	31.3	43.6	12.1	14.73	3.13
13:00	5.10	141	32.4	43.4	12.2	15.06	3.13
13:20	4.72	142	33.7	44.6	11.7	14.28	3.15
13:40	5.04	144	32.2	44.4	12.3	15.23	3.2
14:00	4.71	143	32.5	40.5	11.7	14.14	3.21
14:20	4.60	143	32.6	44.4	11.6	13.97	3.21
14:40	4.70	144	33.0	44.8	11.8	14.39	3.24
15:00	5.04	145	32.8	45.5	12.2	15.11	3.20
15:20	4.83	149	32.8	44.7	11.9	14.54	3.40
average	4.84	137	31.1	42.9	12.08	14.65	2.95

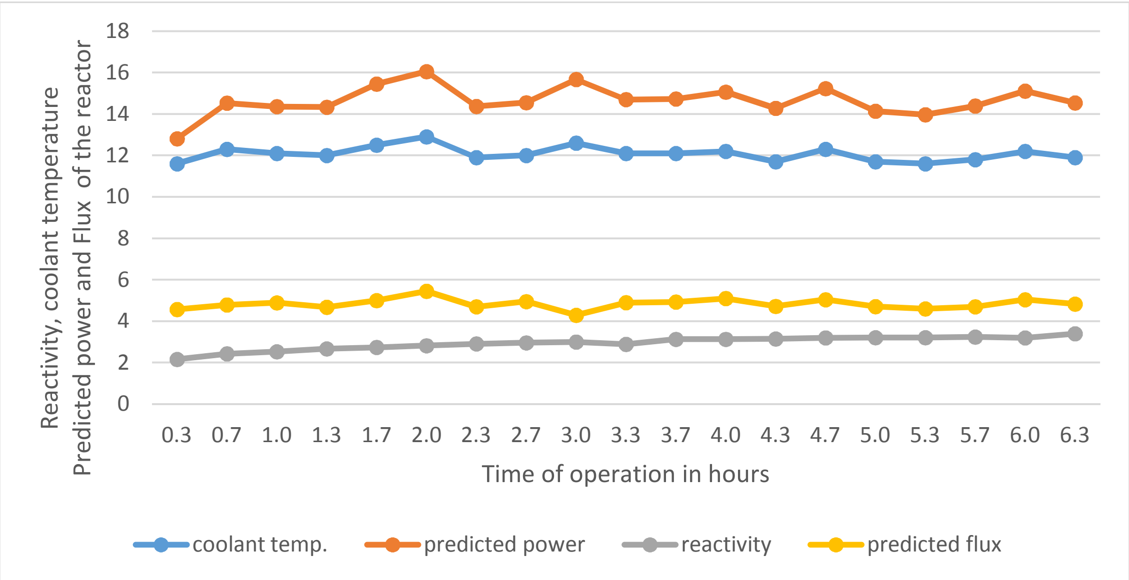

Figure 2: Variation of reactivity, coolant temperature, predicted power and flux with time of operation.

The average value for the predicted power of the reactor at half flux for table 1 and 2 were found to be 14.65 ± 0.35 and 14.65 ± 0.35 kW with average values of coolant temperature of 12.10 °C and 12.08 °C respectively. It was observe that the predicted values were approximately equal even though the values were collected with a month interval. This shows that the installation made on the reactor does not affect power and coolant temperature of the reactor.

Cadmium poison was installed at the Nigeria Research Reactor-1 to increase the reactivity in the system. After years of operation, the reactivity was observed to be decreasing to a value of 2.2 mk (CERT, (2011). This necessitates the removal of cadmium poison in order to increase the excess reactivity of the reactor. 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.

The impact of coolant temperature and control rod withdrawal on the predicted flux which shows an oscillatory behavior have an approximate values of 4.84×10^^{11}n $cm^{-2}s^{-1}$ for the both table. This indicated that there was no rise in the different between average values of the predicted flux and preset values. The Figures 1 and 2 demonstrate that the reactor was not totally cooled during the period of non-operation and that have no critical impact on average predicted flux of the operation which likewise demonstrates the wavering nature of the predicted flux around 5.00×10^^{11}n $cm^{-2}s^{-1}$ at shutdown.

Moreover the coolant temperature difference, predicted power and the reactivity 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 1 and 2.

For the reason of ENAA the cadmium line channel was permanently installed which was achieved via withdrawal of the control rod from the core and the reactor was made critical with the help of cadmium shim capsules because of the High demand for epithermal and thermal neutrons by the clients of the NIRR-1, a MNSR. Keeping the reactor in auto mode, the system reactivity was calculated after half withdrawal of the control rod. After successful installation cadmium line of the reactor, tests were performed to check the safe operation of the reactor. 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.*, 2011). In any case, as time increment, some kind of change is expert between heat generated in the core and heat being lost in light of cooling of the core and incompletely collected in the water of the reactor vessel.

CONCLUSION

Measurements were performed in the NIRR-1 with 347 fuel pins in the core. Our outcome demonstrates that the excess reactivity of the reactor prior to the removal of the cadmium-poison was reported to be 2.2 mk (CERT, 2011). The experiments that was conducted here enable us to achieve an average core excess reactivity of 2.96 mk and 2.95 mk respectively. Due to 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. This brings NIRR-1 to a significant excess reactivity margin without the need of adding beryllium shim as the case could have been or core conversion from high enriched uranium (HEU) to low enriched uranium (LEU). The results of neutron flux measurements show that there are no significant flux variations, coolant temperature and the predicted power of the reactor. It has been demonstrated here that this work will assist tremendously in safe and optimum use of epithermal and thermal neutrons for neutron activation analysis (NAA). Finally, this work shows that the thermal neutron flux of the channel has not significantly deviated from the preset flux measured earlier (Jonah *et al.*, 2006) and the one done recently at the Ghana MNSR (Abrefal *et al.*, 2010).

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