IMPACT OF NIGERIA RESEARCH REACTOR-1 CONVERSION ON ITS THERMAL POWER CALIBRATION

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Abstract: Accurate thermal power calibration is important to the safety and stability of the Nigeria Research Reactor (NIRR-1). The conversion of the NIRR-1 led to the increase in its maximum steady-state thermal power from 31 kW to 34 kW to ensure effective utilization for Neutron Activation Analysis (NAA). In this work, the impact of conversion on the thermal power calibration of the NIRR-1 was assessed using heat balance, calorimetric methods and flux-power relationship. For the calorimetric method, the total power dissipated was 2.3 kW with an uncertainty of ±26%. The total heat loss from the reactor was 1.1 kW. The average pool temperature was 22.70±0.12. For the heat balance method, the value of the average core flow rate, inlet and outlet temperature is 0.215 kg/s, 26.54±0.48 and 30.35±0.64°C, respectively while the total heat loss was 0.0065 kW. The reactor power dissipated was found to be 3.41506 kW, with uncertainty of ±8%. The uncertainty in power at a flux of $1.0 \times 10^{11}\, ncm^{-2}s^{-1}$ which was ± 5% for the HEU core was found to have slightly increased to ± 8% using the heat balance method. Our results confirms that the heat balance method is more accurate in thermal power calibration of the reactor at low power for the LEU core compared to the calorimetric method and the flux-power-relationship is linear but reveals an increase in heat losses and uncertainty in thermal power and an increase in the factor for flux-power-relationship by 9.677%.

Keywords: Power calibration, thermal hydraulics, heat balance, calorimetric, NIRR-1

Introduction

Thermal power calibration of Research Reactors (RRs) is the determination of the actual thermal power of the reactor as a result of the fission reactions taking place in the core of the reactor. It is generally achieved using neutron measuring instruments that measure the neutron flux in the core using fission chambers. The neutron measuring instruments in the NIRR-1 are two fission chambers installed close to the irradiation site (CERT, 2019). The measured neutron flux in the NIRR-1 core is related to the thermal power of the reactor. However, research has shown that the readings from the neutron measuring instruments cannot be solely relied upon because of the difference between preset neutron flux (or power) during operation and that measured by the neutron measuring instruments (fission chambers) installed close to the core (Amponsah *et al*., 2015). Consequently, it is difficult to effectively and reliably monitor or predict the power dissipated in the core. Since additional neutron detectors and measuring instruments can sometimes involve large instrumentation and/or assembly of other devices if they are to be used for power calibration, other instruments such as thermocouples that measure other physical parameters (temperature) related to the thermal power of the reactor are mostly employed for alternative power calibration methods.

The heat balance and the calorimetric methods are alternative power calibration options that have proved to be effective in calibrating and predicting the thermal power of RRs (Mesquita, 2007; Mesquita, 2011; Agbo *et al*., 2015). Both methods rely on heating effects that set in as a result of the thermal power of the reactor during operation. These heating effects are measured by placing thermocouples in strategic positions of the reactor to measure the inlet, outlet and pool temperature of the core. In the NIRR-1, the measurement of the temperature difference between core outlet and inlet is accomplished with two alumel-chromel thermocouples. One is located at the outside of the side beryllium annulus near the core inlet orifice to measure the inlet temperature. The other is located at the upper part of the side beryllium annulus near the core outlet orifice to measure the outlet temperature. Combined, these two pairs of thermocouples can monitor the temperature difference of the reactor coolant. Another set of temperature indicating meters are also dedicated for indicating the reactor water inlet temperature and pool water temperature (CERT, 2019).

Although power calibrations of the NIRR-1 had been done using the heat balance and the calorimetry methods (Agbo *et al.*, 2015); however, these calibrations were done when the reactor was fueled with a High Enriched Uranium (HEU) core. As can be seen in Table 1, the full power of the HEU fueled NIRR-1 was 31 kW. Due to the conversion of the NIRR-1 to UO_2 Low Enriched Uranium (LEU), the full power of the NIRR-1 has been increased to 34 kW to ensure effective utilization of the reactor for Neutron Activation Analysis (NAA) (CERT, 2019).

Table 1: Comparison between NIRR-1 HEU and LEU parameters

Parameter	HEU	LEU
Full power	31 kW	34Kw
No. of active fuel pin/dummy pins	347/3	335/15
Total no. of fuel pins	350	350
Enrichment	90.2%	~13%
Core radius	115 mm	115 mm

This increase in the thermal power of the NIRR-1 LEU core as a result of its conversion unavoidably affects the heat regime in the core due to the fission process in the core. Consequently, previous calibrations cannot be solely relied upon for safe operation of the NIRR-1 LEU core. In the present study, the heat balance and calorimetric methods and flux-to-power relationship were analyzed for the calibration of the NIRR-1 at low power. This is very important in order to investigate the impact of conversion on thermal power calibration methods and enhance safe monitoring, predictability and stability of the reactor.

Materials and Methods

Theory

Calorimetric (slope) method of power calibration

The thermal power of a NIRR-1 and of course other research reactors that are cooled by light water can be determined by measuring the rise in the water temperature over a given period (Bullock, 1965; Zagar *et al.*, 1999). In the calorimetric method, temperature-rise rate (ΔT/Δt) is determined and the reactor power as a function of temperature-rise rate is also determined. This method of power calibration is dependent on heat transfer effects, natural convection of water, heat loss and similar effects. The basic formulation assumes that the reactor power level is calculated by taking the slope of the graph of temperature against time (which is equivalent to the temperature rise rate) and multiplying the slope by the pool constant of the reactor (Zagar *et al.*, 1999; Mesquita *et al.*, 2011):

$$Q = \frac{\Delta T}{\Delta t} K \qquad (1a)$$

Where: Q is the power and k is the experimentally determined heat capacity constant given by

$$k = \rho V_w C_P \qquad (1b)$$

Where: ρ is the density of water, V_w is the volume of the water and C_P is the specific heat capacity of water.

From the calorimetric method equation given by (1a), the power uncertainty, U_q, is calculated using equation (2) and (3) (Mesquita *et al.*, 2007):

$$U'_q = \left\{\left(\frac{\partial q_{u_\rho}}{\partial \rho}\right)^2 + \left(\frac{\partial q_{u_{c_p}}}{\partial c_p}\right)^2 + \left(\frac{\partial q_{u_{v_w}}}{\partial v_w}\right)^2 + \left(\frac{\partial q_{u_T}}{\partial T}\right)^2 + \left(\frac{\partial q_{u_t}}{\partial t}\right)^2\right\}^{\frac{1}{2}} \qquad (2)$$

Where: u_ρ ,u_{c_p}, u_T and u_t are the consolidated uncertainties of the independent primary variables ρ, c_p, v_w, v_w, T and t. Equation (3) is obtained by solving the partial differential equation (2):

$$\frac{U_q}{q} = \left\{\left(\frac{U_\rho}{\rho}\right)^2 + \left(\frac{U_{C_p}}{C_P}\right)^2 + \left(\frac{U_{v_w}}{V_w}\right)^2 + \left(\frac{U_T}{T}\right)^2 + \left(\frac{U_t}{t}\right)^2\right\}^{\frac{1}{2}} \qquad (3)$$

Power calibration by heat balance method

The heat balance method assumes that the heat generation rate in the core is related to the coolant temperature difference across the core and the flow rate of the coolant passing through the core. This is expressed mathematically in the form of equation (4a) (Agbo *et al.*, 2015; Mesquita *et al.*, 2011).

$$Q \propto \dot{m}\Delta T \qquad (4a)$$

When the proportionality sign in 4a is removed, one gets:

$$Q = C_p \dot{m} \Delta T \qquad (4b)$$

Where: Q is the thermal power dissipated rate (heat generation rate kW), ṁ is the flow rate of coolant passing through the core (kg/s), C_p is the specific heat capacity of a coolant (kj/kg◦C) and ΔT, which also expressed as (T_{out} - T_{in}), is the coolant temperature difference (°C). The flow rate can be measured by an orifice plate and a differential pressure transmitter. The NIRR-1 does not have an installed device for measuring the flow rate. However, one can determine the flow rate using equation (5).

$$\dot{m} = \frac{\rho V}{\Delta t} \qquad (5)$$

Where: ρ is the density of the coolant (kg/m^3), V is the volume of the coolant passing through the core (m^3), and Δt is the change in time (sec). The coolant temperature and density of the coolant passing through the core of NIRR-1 is related by equation (6) (Mansir *et al.*, 2012):

$$\rho = 2E - 05k^3 - 0.006k^2 - 0.0233k - 999.97 \qquad (6)$$

Where: ρ is the coolant density (kg/m^3) and k is the coolant temperature rise (◦C).

The power uncertainty U_q is given by the following (Bullock, 1965; Holman, 1998):

$$U_q = \left\{\left(\frac{\partial q_{u_{\dot{m}}}}{\partial \dot{m}}\right)^2 + \left(\frac{\partial q_{u_{C_p}}}{\partial C_p}\right)^2 + \left(\frac{\partial q_{u_{T_{int}}}}{\partial T_{int}}\right)^2 + \left(\frac{\partial q_{u_{T_{out}}}}{\partial T_{out}}\right)^2\right\}^{\frac{1}{2}} \qquad (7)$$

Where: $U_{\dot{m}}$, U_{C_p}, $U_{T_{int}} and\ U_{T_{out}}$, are the consolidated uncertainties of the independent primary variables ṁC_P, V_w, T and t. Equation (8) is obtained by solving the partial differential equation (7):

$$\frac{U_q}{q} = \left\{\left(\frac{U_{\dot{m}}}{\dot{m}}\right)^2 + \left(\frac{U_{C_P}}{C_P}\right)^2 + \left(\frac{U_{T_{int}}}{T_{int}}\right)^2 + \left(\frac{U_{T_{out}}}{T_{0ut}}\right)^2\right\}^{\frac{1}{2}} \qquad (8)$$

Heat losses from the reactor pool to the environment

The core of the NIRR-1 is placed below the room floor in the bottom of a cylindrical pool. The design of the pool matches industrial building standards. The heat transfer of the reactor pool is through conduction; convection and evaporation (Fig. 1).

Heat loss by conduction

The heat generated in the fuel of NIRR-1is transferred to the water through the natural convection from the water of the reactor to the reinforced concrete lined with stainless steel by conduction and is finally cooled by the ambient air through convection (Fig. 1). To model the heat from the core to the outside surface, a thermal circuit of the reactor is created (Cengel and Boles, 2008). First Fourier's law of heat transfer is expressed in cylindrical form as (Cengel and Boles, 2008):

$$q_r = k(2\pi r L)\frac{dT}{dr} \qquad (9)$$

Where: q is the heat rate (W), k is the thermal conductivity (W/m-k), r is the radial distance (m), L is the cylindrical length (m), and T is the temperature (K).

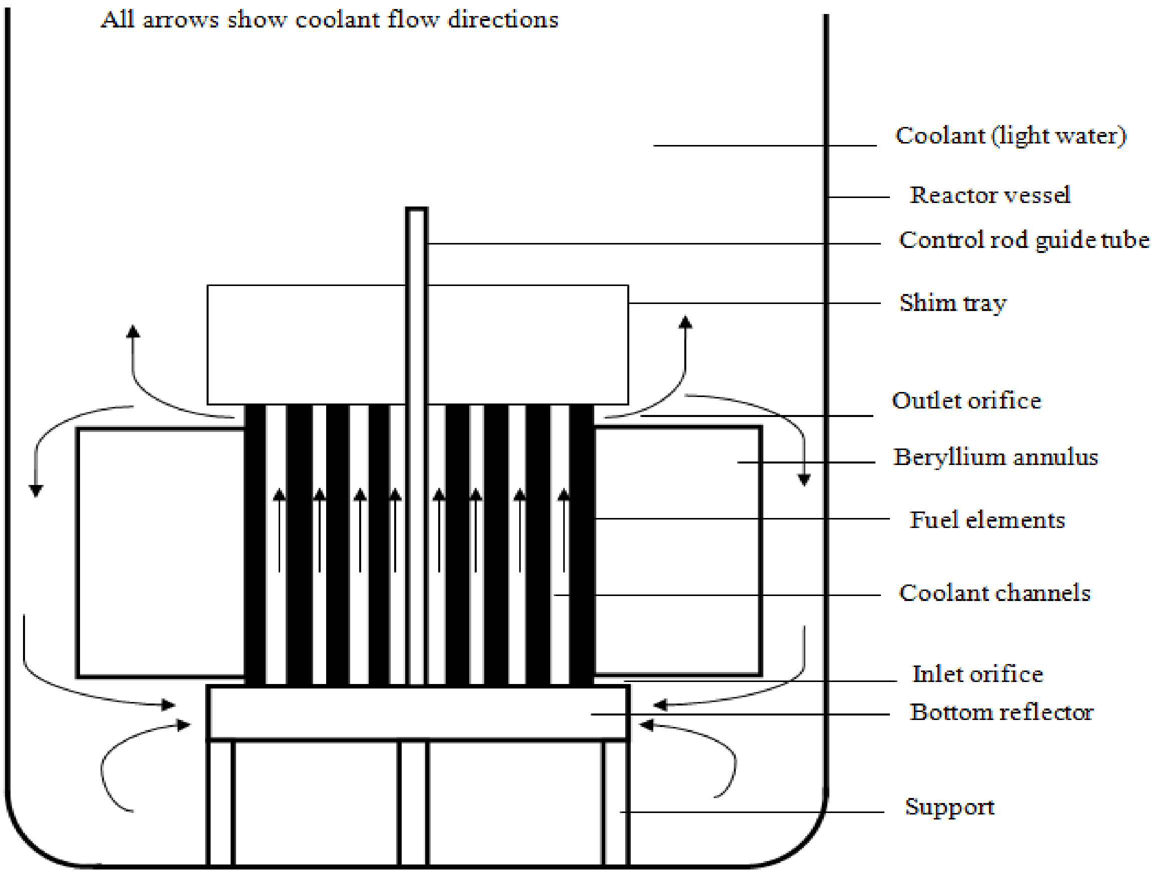

Fig 1: Schematic diagram for the Natural Circulation Coolant Flow Pattern in NIRR-1 (CERT, 2019)

Applying the general solution to equation (9) and using the temperatures of the inner and outer surfaces as boundary conditions creates an expression for the heat transfer rate H_t;

$$H_t = \frac{2\pi Lk\left(T_{s,1} - T_{s,2}\right)}{In\frac{r_2}{r_1}} \qquad (10)$$

When using the thermal circuit model, the material properties and dimensions are separated out of equation (9) to calculate total thermal resistance. Equation (11) is the cylindrical thermal resistance for conduction.

$$R_{conc} = \frac{In\frac{r_2}{r_1}}{2\pi Lk} \qquad (11)$$

Where: R is the thermal resistance (k/W).

The insulating materials in NIRR-1 are arrayed in a serial configuration so that the thermal circuit takes the form of equation (12). The heat transfer to the outer surface of the steel tank is then given knowing the temperature of the core in contact with the water and temperature of the outer stainless steel tank (Incropera *et al*., 2007):

$$q_r = \frac{T_{core} - T_{ss}}{\left(\frac{1}{2\pi r_{core} L_{core} h}\right) + 2\left(\frac{In\frac{r_{ss,2}}{r_{ss,1}}}{2\pi k_{ss} L_{ss}}\right) + \left(\frac{In\frac{r_{conc,2}}{r_{conc,1}}}{2\pi k_{conc} L_{conc}}\right)} \qquad (12)$$

Where: T_{core} is temperature of the core (in contact with water), T_{ss} is temperature of the outer surface steel tank, r_{core} is core radius, L_{core} is core height, h is heat transfer coefficient of water, $r_{conc,1}$ is inner radius of concrete layer, $r_{conc,2}$ is outer radius of the of concrete layer,k_{conc} is thermal conductivity of concrete, $r_{ss,1}$ is inner radius of the stainless steel, $r_{ss,2}$ is outer radius of stsinless steel tank, k_{ss} is thermal conductivity of stainless steel, L_{ss} is height of stainless steel tank.

Equation (12) can only be used to find the heat loss through the tank walls if the convective heat transfer coefficient (h) of the tank water is known. The Grashof number is given by (Holman, 2002; Zagar *et al.*, 1999);

$$Gr = \frac{g\beta(T_{core} - T_O)L_{core}^3}{V^2} \qquad (13)$$

Where: Gr is Grashof number, g is gravitational acceleration ($9.8^{m}/_{s^2}$), To is bulk pool temperature (20°C), β is volumetric thermal expansion coefficient of water ($207.71\times 10^{-6}k^{-1}at$ 20°C); V is kinetic viscosity of water ($1.0058\times 10^{-6}m^2s^{-1}$, at 20°C), (Cengel and Boles, 2005); and l is characteristics length of the heat transfer surface, equivalent to 0.9 times the diameter of the pool (i.e. 2.43 m).

Multiplying equation (13) by the Prandtl number, which is 7 for water at 20℃, gives the Rayleigh number given by Zagar *et al.* (1999), Incropera *et al*. (2007):

$$Nu = 0.15Ra_l^{\frac{1}{3}} \qquad (14)$$

The relationship between transfer coefficient (h) the Nusselt number (Nu) is derived from Newton's law of cooling and is given by;

$$Nu_l = \frac{hl}{k} \qquad (15)$$

Where: k= thermal conductivity of water at 20℃ = 5.984×10^{-3} W/m (Cengel and Boles, 2008)

Heat loss by evaporation

The heat loss due to the evaporation in the upper surface of the reactor pool is calculated using equation (16) (Holman, 2002):

$$q_e = \dot{n}\gamma \qquad (16a)$$

Where: γ is the difference between the specific enthalpy of saturated water and the specific enthalpy of saturated steam at wet –bulb temperature of the air in the reactor room, and ṅ is the rate of mass transfer from the pool to the air given by the equation;

$$\dot{n} = h_d A \rho_{air} (C_{sat-} C_{\infty}) \quad (16b)$$

Where: A is the area of the upper surface of the reactor pool, ρ_{air} is the airdensity, C_{sat} is the vapor concentration at saturated conditions for the air at the reactor room temperature, C_{∞} is the vapor concentration in the air in the reactor room, and h_d is the mass transfer coefficient given by;

$$h_d = \frac{h_c}{\rho_{air} C_{p_{air}}} \left(\frac{p_r}{S_c}\right)^{\frac{2}{3}} \quad (16c)$$

Where: p_r is the prandtl number (0.713 for the air at 20℃), S_c is the Schmidt number (0.612 for water vapor diffusing in the air at 20℃), $C_{p_{air}}$ is the heat capacity of the air, h_c is the convection heat transfer coefficient obtained in terms of Nusselt number in equation (15), where k is the thermal conductivity in the air (0.0257 W/m-K) (Cengel and Boles, 2008). The Nusselt number can be expressed as;

$$Nu = 0.14(G_r \times P_r)^{\frac{1}{3}} \quad (17)$$

Where: the Grashof number Gr, is expressed in the form of equation (18)

$$Gr = \frac{g\beta(T_{sur} - T_{\infty})L^3}{V^2} \quad (18)$$

Where: g is gravitational acceleration (9.8 m/s^2), β is the volumetric thermal expansion coefficient of air ($207.71 \times 10^{-6} k^{-1}$ at 20℃, V is kinetic viscosity of air ($15.11 \times 10^{-6}\, m^2 s^{-1}\, at$ 20℃) (Cengel and Boles, 2008), T_{sur} is the water pool temperature at the surface and T_{∞} is the air temperature in the reactor room.

Heat loss by convection

Newton's law is used in this work to estimate the rate of heat convection from the pool surface to the air (Zagar *et al*, 1999; Incropera *et al*., 2007):

$$q'' = h.(T_O - T_{air}) \quad (19)$$

The Rayleigh number is in this case is expressed as follows;

$$R_{a_l} = \frac{g\beta(T_0 - T_{air})L^3}{\propto V^2} \quad (20)$$

Where: g is the acceleration due to gravity (9.8 m/s^2), β is the volumetric thermal expansion coefficient of the air ($3.4 \times 10^{-3} K^{-1} at$ 20℃,), T_O is the bulk pool temperature, T_{air} is the air temperature in the reactor room, and v is the kinetic viscosity of the air ($15.11 \times 10^{-6}\, {m^2}/{s}\, at$ 20℃), and $\propto$ is air thermal diffusivity ($1.9 \times 10^{-5}\, {m^2}/{s}$) (Cengel and Boles, 2008).

Flox-to-power relationship

The equation that relates the thermal power, macroscopic fission cross section, core volume and thermal neutron flux of the NIRR-1 is (Ahmed, 2006):

$$P = 3.1\text{x}10^{-10} \textstyle\sum_f V_f \phi \quad (21)$$

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) = 23 cm; Core radius (r) = 11.5 cm; $\sum_f$ = macroscopic fission cross-section of the core fuel

Since all the parameters on the right hand side of equation (21) are constant except the neutron flux ϕ, the flu-to-power relationship for the HEU core is reduced to (Ahmed, 2006: CERT, 2011):

$$P = 3.1\text{x}10^{-8}\phi \quad (22)$$

Experimental procedure

The reactor power was preset to 3.4 kW corresponding to neutron flux of $1.0 \times 10^{11}\, ncm^{-2}s^{-1}$ following the normal start-up procedure. All the sources of heat added to the pool water, specifically the pool lights, were turned off. The bulk pool temperature (T_0) and the temperature of air (T_{air}) in the reactor room was measured and recorded before the reactor startup. High precision temperature detector (thermocouple) was employed to measure inlet, outlet and pool temperatures at 6 minutes interval for about 2 hours steady-state operation. The inlet temperature, outlet temperature, and pool temperature were recorded simultaneously. No sample was irradiated during the experiment to avoid reactivity effects induced by sample irradiation.

Results and Discussion

Tables 2 and 4 showed the results of the calorimetric and heat balance calibration, respectively. The reactor parameters evaluated using data obtained from Tables 2 and 4 are presented in Tables 3 and 5, respectively. The plotted graph of pool temperature against time is presented in Figs. 2. Table 6 shows the comparison between the uncertainties values obtained in this work and other calibrations done in previous work for the NIRR-1 HEU core.

Table 2: Data Obtained for Calibration at Low Power (3.4 kW) by Calorimetric (slope) method.

Time (h)	Tin (ºC)	Tout (ºC)	To (ºC)
0.00	25.90	29.20	22.40
0.10	26.00	29.50	22.40
0.20	26.00	29.60	22.50
0.30	26.10	29.70	22.70
0.40	26.20	30.30	22.60
0.50	26.00	30.00	22.60
0.60	26.20	30.40	22.70
0.70	26.10	29.90	22.80
0.80	26.40	30.40	22.80
0.90	26.30	30.80	22.70
1.00	26.90	30.60	22.70
1.10	26.80	30.40	22.80
1.20	27.00	30.60	22.80
1.30	27.10	30.70	22.70
1.40	27.30	30.80	22.80
1.50	27.00	31.30	22.70
1.60	26.90	31.10	22.90
1.70	27.00	30.40	22.80
1.80	27.10	30.90	22.90

Table 3: Calculated parameters from data obtained from Table 2

Parameter	Value
Temperature rise rate ΔT/Δt (ºC/h)	0.19
Average time (h)	0.9±0.56
temperature rise range (ºC)	22.4-22.9
Average pool temperature (ºC)	22.70±0.12
Power dissipated (kW)	2.3
Thermal losses from the reactor pool (kW)	1.1
Total reactor power (kW)	3.4
Uncertainty (kW)	0.6(26%)

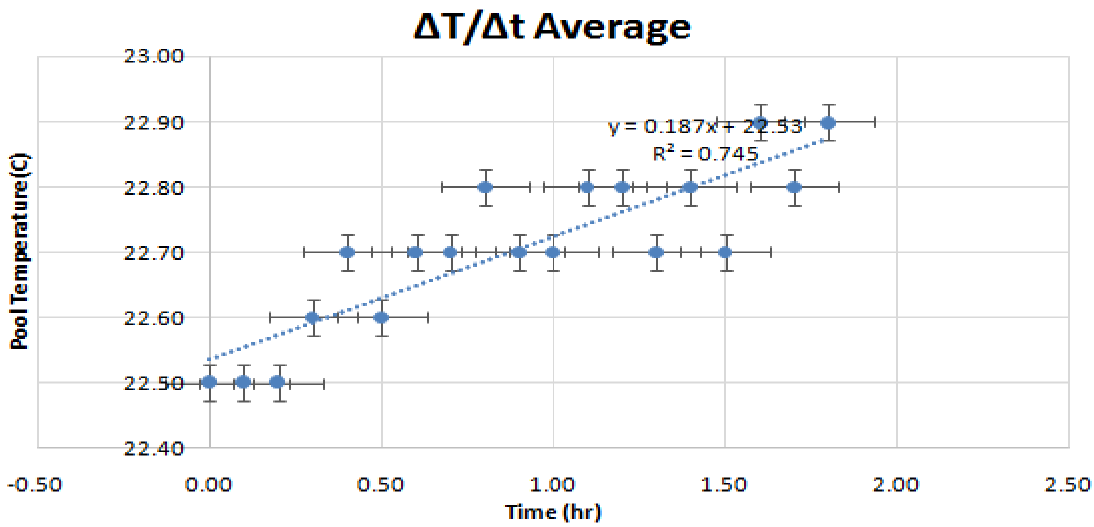

Fig. 2: Variation of pool temperature with operation time

Table 4: Data obtained for calibration at low power (3.4 kW) by heat balance method

Time (h)	Tin (ºC)	Tout (ºC)	ΔT (˚C)	ρ (kg/m^3)	$\dot{m}$ (kg/s)	Measured Flux ($ncm^{-2}s^{-1}$)
0	25.9	29.2	3.3	999.982	0.246	1.00E+11
0.1	26	29.5	3.5	999.979	0.232	1.00E+11
0.2	26	29.6	3.6	999.977	0.226	1.09E+11
0.3	26.1	29.7	3.6	999.977	0.226	1.00E+11
0.4	26.2	30.3	2	999.966	0.198	1.01E+11
0.5	26	30	4	999.968	0.203	1.00E+11
0.6	26.2	30.4	3	999.964	0.194	1.00E+11
0.7	26.1	29.9	3.8	999.973	0.214	9.98E+10
0.8	26.4	30.4	4	999.968	0.203	9.97E+10
0.9	26.3	30.8	4.5	999.955	0.181	1.00E+11
1	26.9	30.6	3.7	999.975	0.22	1.00E+11
1.1	26.8	30.4	3.6	999.977	0.226	1.00E+11
1.2	27	30.6	3.6	999.977	0.226	9.98E+10
1.3	27.1	31.7	3.6	999.977	0.226	1.01E+11
1.4	27.3	30.8	3.5	999.979	0.232	1.00E+11
1.5	27	31.3	4	999.961	0.189	1.00E+11
1.6	26.9	31.1	3	999.964	0.194	1.00E+11
1.7	27	30.4	3.4	999.981	0.239	1.00E+11
1.8	27.1	30.9	3.8	999.973	0.214	1.00E+11

Table 5: Calculated parameters from data obtained from Table 4

Parameter	Value
Average coolant flow rate (kg/s)	0.215±0.018
Average inlet temperature (ºC)	26.54±0.48
Average outlet temperature (ºC)	30.35±0.64
Average ΔT (ºC)	3.80±0.34
Average flux($ncm^{-2}s^{-1}$)	1.006E+11
Power dissipated (kW)	3.415060
Total heat losses from the reactor pool (kW)	0.0065
Total reactor power (kW)	3.421581
Uncertainty	0.29 (±8%)
% deviation of power dissipated from total reactor power	0.19%

Table 6: Comparison between the uncertainties obtained in this work and other calibrations at low power

Reactor name	Calibration method	Uncertainty value (%)	Reference
NIRR-1 HEU core	Heat balance	5	Agbo *et al*. (2015)
NIRR-1 LEU core	Heat balance	8	This work
NIRR-1 HEU core	Calorimetric	-	
NIRR-1 LEU core	Calorimetric	26	This work
NIRR-1 HEU core	Dose rate	9.7	Yahaya *et al*. (2016)

Calorimetric method at low power (3.4 kW)

The result as presented in Table 2 shows that there is a steady rise in pool temperature with time. The average pool temperature is 22.70±0.12. The temperature rise range is 22.4 – 22.9ºC. It can be observed from Fig. 2 that the graph has a poor-fitting with a least square regression line because some of the fitted points show a significant deviation from this line. This line is therefore not a best fit but a close fit. The value of the regression coefficient (R^2) is 0.6759. This value implies a fairly good but not very accurate correlation. When the total heat losses obtained is added to the power dissipated the total thermal power of 3.4 kW with an uncertainty of ±26% as seen in Table 3. Because the correlation coefficient in Fig. 2 is not strong and the linear fit which the accuracy of the calorimetric (slope method) depend upon is poor, the accuracy of the result of power calibration using this method is unreliable at low power. Consequently, this method of power calibration is not suitable for calibration at low power for the NIRR-1 LEU core. The failure of this method in accurately calibrating the thermal power of the LEU core at low power is in agreement with a similar study carried out at low power of 3.6 kW for the NIRR-1 HEU core (Agbo *et al*., 2015).

Heat balance method calibration at low power (3.4 kW)

As can be seen in Table 4, the value of the average core flow rate, inlet and outlet temperature is 0.215 kg/s, 26.54±0.48 and 30.35±0.64ºC, respectively. The core coolant temperature is ≤ 4ºC. A steady but not significant rise in inlet and outlet temperatures with time can be observed in Table 4. This is due to sufficient thermal circulation of coolant in the core at this power. The total thermal power calculated is 3.421581 kW. The calculated heat losses was found to be 0.0065 kW. This implies that at a low power of 3.4 kW there is essentially insignificant heat loss from the core of the reactor. The steady but insignificant rise in inlet and outlet temperatures and negligible heat loss from the core is an indication that the reactor is very stable at this power. It can be seen from Table 6 that the uncertainty value of ±8% obtained in this work is comparable to the value of ±5% obtained by Agbo *et al.* (2015) using heat balance method at low power of 3.6 kW for the HEU core. This confirms the relative accuracy of this method at low power for the NIRR-1. Consequently, similar to the HEU core, thermal power calibration for the NIRR-1 LEU core using the heat balance method is more reliable when compared to the calorimetric method at low power.

Flux-to-power relationship

The flux-to-power relationship of the HEU core was given by equation (22). Although the radius and height of the NIRR-1 LEU core is exactly the same as that of the HEU core as seen in Table 1, the flux-to-power relationship of the HEU core is not correct for the LEU core. This is because substituting the measured average flux of 1.006E+11 in our work into equation (22) result in a power of 3.1 kW instead of 3.4 kW. In order to obtain our thermal power of 3.4 kW, equation (22) must become:

$$P = 3.4 \times 10^{-8}\phi \quad (23)$$

The change in the factor relating the thermal power of the NIRR-1 core from 3.1×10^{-8} in equation (22) for the HEU core to 3.4×10^{-8} in equation (23) for the LEU core can be attributed to an increase in the macroscopic fission cross section of ^{235}U due to steady-state full power increase from 31 to 34 kW. The percentage increase in the factor relating reactor power to flux is 9.677%. Equation (23) indicates a linear relationship between the reactor power and the measured neutron flux of the NIRR-1 LEU core which is similar to that of the HEU (Ahmed *et al.*, 2006). This implies that at a measured neutron flux of $5 \times 10^{11} n/cm^2 s$, the reactor power is 17 kW as expected for half power operation of the NIRR-1.

Conclusion

The knowledge of the actual reactor thermal power is very important for precise neutron flux and fuel element burn-up calculation which is important in predicting the isotopic changes in the fuel element of the core. The thermal power of NIRR-1 LEU core was calibrated in this work using the calorimetric (slope) and heat balance method and flux-to-power relationship at low power (3.4 kW) to assess the impact of conversion on thermal power calibration of the reactor. The calorimetric method presented a larger uncertainty of ±26% compared to that of the heat balance method of ±8%. The ±8% uncertainty in thermal power for the heat balance method for the present LEU core is slightly greater than the ±5% uncertainty in power obtained for the HEU core (Agbo *et al.*, 2015). The total heat loss from the reactor using the heat balance method was 0.0065 kW as compared to 1.1 kW total heat loss obtained using the calorimetric method. The heat loss in the core of 0.0065 kW using the heat balance method for the present LEU core is negligible. This is similar to the value of 0.001 kW obtained (Agbo *et al.*, 2015) using the same method for the HEU core at low power. The factor in the linear relationship between thermal power and neutron flux of the NIRR-1 increased from 3.1×10^{-8} for HEU to 3.4×10^{-8} for LEU. This is an increase of 9.677%. Similar to the results of thermal power calibration for the HEU core in previous study (Agbo *et al*, 2015), our results confirms that the heat balance method is more accurate in thermal power calibration of the reactor at low power for the LEU core compared to the calorimetric method and the flux-power-relationship is linear but reveals an increase in heat losses and uncertainty in thermal power and an increase in the factor for flux-power-relationship. This can be attributed to the increase in thermal power from 3.1 to 3.4 kW at the same flux of $1.0 \times 10^{11} ncm^{-2}s^{-1}$ for HEU and LEU respectively due to conversion.

Conflict of Interest

Authors declare that there is no conflict of interest reported in this work.

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