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Transient Studies of Ghana Research Reactor-1 after Nineteen (19) Years of Operation Using PARET/ANL Code

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Abstract

The Program for the Analysis of Reactor Transients/Argonne National Laboratory (PARET/ANL) code was used to predict the thermal hydraulic behaviour of the Ghana Research Reactor-1 after adding 9.0 mm of beryllium to the top shim tray of the core. The core was analysed for reactivity insertions 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.7 mk, respectively. The reactor is still safe to operate in the range 2.1 mk to 4.0 mk. However, 2.1 mk would be ideal since the reactor automatic shutdown (SCRAM) is set not to exceed 120% of reactor nominal power.

Keywords

Reactivity Insertion, Transients, Control Rod Worth, Power Peaking Factor, Moderator Reactivity Coefficient, Neutronic Parameters

1. Introduction

As a reactor operates, fuel burnup continues with accumulation of fission products in the fuel meat which makes the neutron spectrum softer. Some of the fission products have a high affinity for neutrons which adversely affects the neutron population and proportionally, reactor power. Fission products that absorb neutrons are called neutron poisons. The buildup of fission products in the fuel makes reactivity coefficients more negative [1]. Reactor cores surrounded by beryllium metal might have provision to increase the thickness of beryllium to compensate for neutron population lost due to fuel depletion and accumulation of neutron poisons. Each time changes are made to the reactor core either through refueling or addition of beryllium; neutronics and thermal-hydraulic properties of the reactor are perturbed. Any activity that may influence neutronic, thermal-hydraulic and mechanical properties of a reactor should be supported by safety evaluations. This is to ascertain that the reactor is operating within prescribed safety margins [2].

The Ghana Research Reactor-1 (GHARR-1) has been in operation for nineteen years and due to accumulation of fission products in the fuel meat, the excess reactivity dropped from 4.0 mk to about 2.3 mk. A 9.0 mm layer of beryllium has been added to the top shim tray of the core to reflect more neutrons into the core restoring the excess reactivity to about 4.0 mk. **Figure 1** shows the current schematic vertical cross section of the reactor core with a 9.0 mm beryllium shim added to the top shim tray. In this work, transients of GHARR-1 are evaluated relative to addition of 9.0 mm of beryllium based on the PARET/ANL code.

2. Theory

The change in neutron flux associated with variation in material or geometry of a reactor is accounted for as a reactor transient. This type of transient is bound to occur during normal operations due to control rod movement or during reactivity change. Transient responses are more critical when an accident occur as reactor properties change rapidly with large amplitudes. These conditions require a thorough understanding for improved identification, prevention and mitigation of transients [3]. Transients have an effect on reactor power and there is need to understand the feedback effects of transients. Critical among feedback effects is the interrelation between component temperature and the reactor power.

The study of the dependence of reactor power on temperature through various simulation codes has shown a semi-empirical relationship between coolant inlet temperature, the increase in coolant temperature and reactor power. Equation (1) shows the semi-empirical relationship [4]:

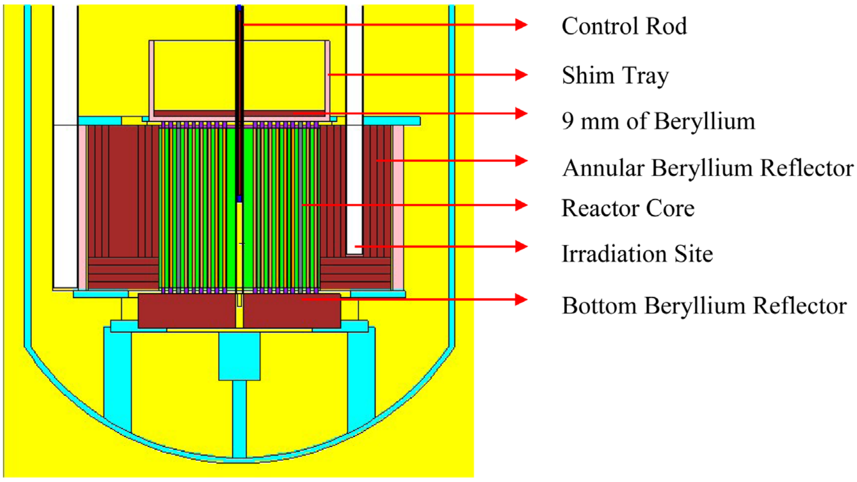

Figure 1. A schematic diagram of the vertical cross-section of GHARR-1.

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

where: ΔT = temperature difference between inlet and outlet orifices (˚C)

H = height of the inlet orifice in millimeters

T_i = inlet temperature (˚C)

P = reactor power level (kW)

For GHARR-1, the orifice height H is kept at 6 mm to keep the reactor safe. Thus substituting 6 mm for H in Equation (1) reduces it to Equation (2)

$$\Delta T = 7.027T_i^{-0.35}P^{(0.59+0.0019T_i)} \tag{2}$$

Introducing the natural log and the exponential operators, Equation (2) can be expressed in terms of power as:

$$P = \exp\left[\left(\ln\frac{\Delta T}{7.027T_i^{-0.35}}\right)\left(0.59+0.0019T_i\right)^{-1}\right] \tag{3}$$

From Equation (3), the relationship between reactor power and coolant temperature is linear. This implies an increase in reactor power would cause a feedback effect of an increase in coolant temperature. This feedback effect makes it possible to predict reactor power using thermal hydraulic parameters.

The GHARR-1 is a miniature neutron source reactor with a small core surrounded by an annular and radial block of beryllium. Beryllium blocks act as neutron reflectors to reduce neutron leakage and conserve the neutron population [5]. The reactor is designed to have an excess reactivity of 4.0 mk. However, due to fuel depletion and accumulation of fission products and in particular neutron poisons, the excess reactivity of the reactor drops from 4.0 mk to an allowed lower limit of 2.3 mk in about 2 years [6]. A layer of beryllium is added to the top shim tray to compensate for loss in excess reactivity.

Like any other nuclear reactor, any changes made to the core should be supported by safety parameter evaluations [7].

3. Experiment

The Reactor Burn up System (REBUS), Monte Carlo N-Particle code version 5 (MCNP5), Program for analysis of Reactor Transients (PARET) codes have been used in this work.

The REBUS-PC computer code provides reactor physics and core design information such as neutron flux distributions in space, energy, and time, and to track isotopic changes in the fuel and neutron absorbers relative to fuel burnup [8]. An inventory of isotopes in the fuel after 19 years of operation was generated using REBUS code. The model was simulated using the current operating scheme of GHARR-1 of 2.5 hours per day, 4 days a week and 52 weeks in a year. The isotopes generated were used to update the material card in the MCNP model of the reactor core. The material card contains information on the elemental and isotopic composition of the reactor core.

The MCNP code has been in development by Los Alamos National Laboratory since 1957 with several further major improvements. It is primarily used for simulation of nuclear processes, such as fission, but has the capability to simulate particle interactions involving neutrons, photons, and electrons [9].

In this work MCNP5 simulations were used to generate the neutronic parameters and the power peaking factors of GHARR-1 after 19 years of operation. Table 1 shows the maximum core power peaking factor, moderator reactivity coefficient and neutronic parameters that were used to update the PARET/ANL model of GHARR-1.

The PARET/ANL Code was initially developed for analysis of the SPERT-III experiments for temperatures and pressures typical of power reactors [10]. Modifications have been made to the code to incorporate reactor thermal–hydraulic analysis; these include departure from nucleate boiling, flow instability, single and two-phase heat transfer correlations, and flow rates encountered in research reactors. It also gives an estimate of voiding produced by subcooled boiling through its optional voiding model. The code has the capability to provide a coupled thermal-hydraulic and point kinetics with continuous reactivity feedback [10] [11].

The conservation equations used in the model are given by Equation (4), (5) and (6) expressed as follows:

$$\frac{\partial \bar{\rho}}{\partial t} = \frac{\partial G}{\partial z} \tag{4}$$

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial t}\left(\frac{1G^2}{\rho'}\right) = \frac{\partial P}{\partial z} - \left(\frac{f}{\rho}\right)\left(\frac{|G|G^2}{2D_e}\right) - \bar{\rho} g \tag{5}$$

$$\rho'' \frac{\partial E}{\partial t} + G\frac{\partial E}{\partial z} = q \tag{6}$$

where $\bar{\rho} = \rho_l(1-\alpha) + \rho_v$ is the average density

$\frac{1}{\rho''} = \left(\frac{(1-\chi)^2}{\rho_l(1-\alpha)}\right) + \frac{\chi^2}{\rho_v \alpha}$ is the momentum density

$\rho'' = \left[\rho_l \chi + \rho_v(1+\chi)\right]\frac{\partial \alpha}{\partial \chi}$ is the slip flow density

$\rho_v, \rho_l, \chi, \alpha$ are: saturated vapor and liquid densities, vapor weight fraction and vapor volume fraction respectively, G is mass flow rate, P is pressure; E is

Table 1. Neutronic and Kinetic Parameters of GHARR-1 after 19 years of operation.

Parameter	Value
Excess reactivity (mk)	3.86
Control rod worth	6.98
Delayed neutron fraction $(\Delta k k^{-1})$	8.17507×10^{-3}
Neutron generation time $(\Delta k k^{-1})$	8.147×10^{-5}
Moderator reactivity coefficient	−0.1218
Maximum power peaking factor	1.352

enthalpy, f is friction factor, g is gravitational acceleration, and q = heat.

The updated core model was simulated for slow transients with reactivity insertions of 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.7 mk, respectively. These values were chosen based on the control rod worth of the reactor reported as 6.8 mk in the SAR of GHARR-1. Control rod worth is an important parameter in the design and analysis of a nuclear core. It was imperative to cover the reactivity range of the control rod worth in order to have thermal-hydraulic analyses that were reflective of the reactor control system. These values were also chosen for comparison's sake as they are the ones quoted in the SAR and other literature on GHARR-1.

4. Results and Discussion

Table 2 shows a comparison of feedback effects predicted in this work and those reported in the SAR of GHARR-1, for reactivity insertion of 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.7 mk, respectively. Results predicted from this work are in agreement with experimental data reported in the SAR. However, PARET/ANL code was unable to simulate correctly reactivity insertions for 5.0 mk. This is observed through the huge difference between experimental data and that predicted using PARET/ANL code for reactivity insertion of 5.0 mk. This limitation by PARET/ANL (Version 7.3 of 2007) code to simulate higher reactivity insertion can be attributed to its inability to switch within thermal-hydraulic flow regimes [11] [12].

Results presented in Table 2 are plotted in Figures 2-6. Figure 2 shows a comparison in reactor power for reactivity insertions 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.71 mk, respectively. For reactivity insertions of 5.0 mk and 6.7 mk, reactor power rises sharply within seconds. This sharp rise in power within a short time explains the behaviour of a reactor during an accidental insertion of high reactivity. However, for lower reactivity insertions, the rise in power is gradual. The 6.7 mk reactivity insertion has a sharp rise in power and attains its maximum power in 600 s. The 2.1 mk and 3.0 mk reactivity insertion have a gradual rise in power but attain their maximum power in 612 s and 606 s,

Table 2. Comparison of reactivity feedback between this work and experimental data.

Reactivity Insertion (mk)	Maximum Power (kW)		Fuel Temperature (˚C)		Clad surface temperature (˚C)		Coolant outlet temperature (˚C)	
	Exp. SAR	This work	Exp. SAR	This work	Exp. SAR	This work	Exp. SAR	This work
2.1	36 ± 4	39.3	66 ± 3	64.1	-	63.5	47 ± 4	51.6
3.0	-	55.3	83 ± 2	74.2	82 ± 5	73.3	57 ± 2	57.8
4.0	100 ± 5	91.2	-	93.6	-	92.2	72 ± 3	69.6
5.0	60 ± 7	130	-	111	-	109	-	79.9
6.7	-	202	-	137	-	134	-	96.3

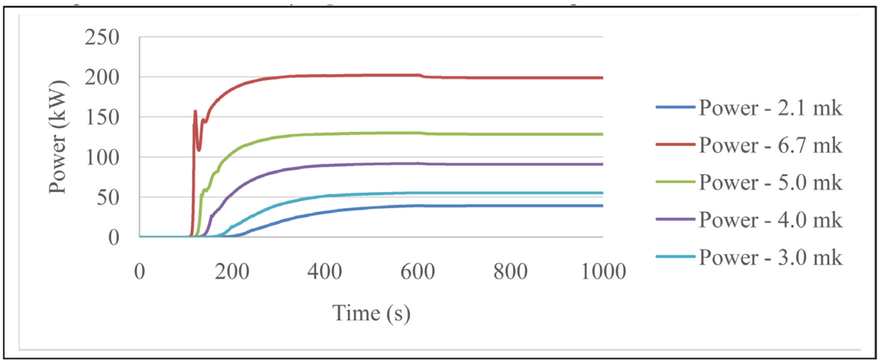

Figure 2. Plot of reactor power for 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.7 mk reactivity insertions.

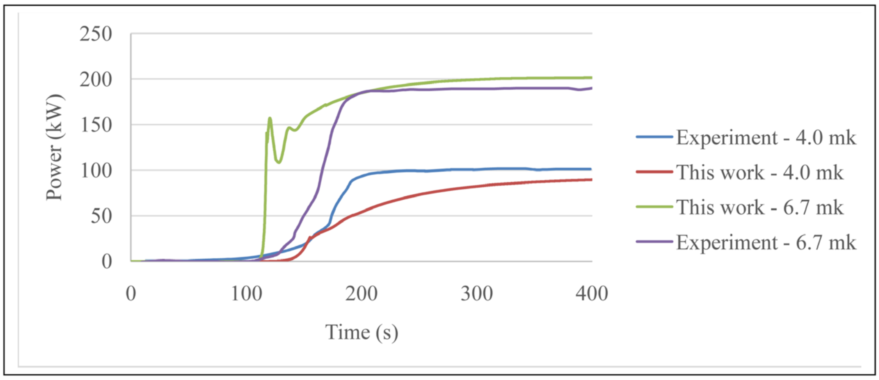

Figure 3. Comparison of reactor power (kW) for 4.0 mk and 6.71 mk reactivity insertion and experimental work (Akaho and Maakuu, 2002).

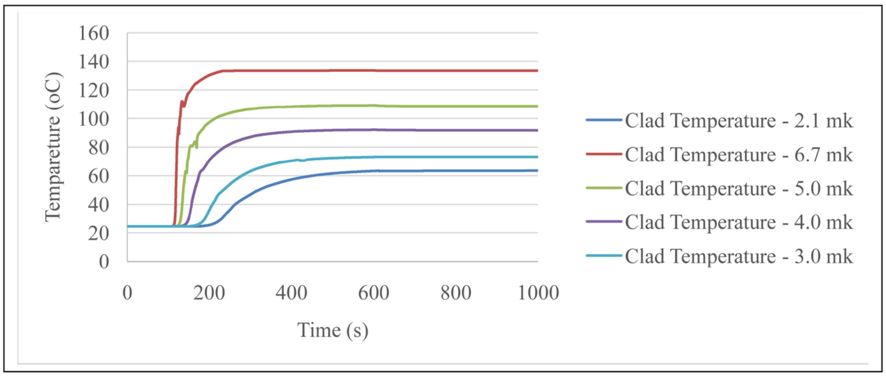

Figure 4. Plot of Clad temperature for 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.71 mk reactivity insertions.

respectively. For reactivity insertions of 4.0 mk and 5.0 mk, the maximum power is attained at 144.36 s and 144.06 s, respectively. The noise observed in reactor power plots for reactivity insertion 5.0 mk and 6.7 mk amplify PARET/ANL code's inability to switch within thermal-hydraulic flow regimes for higher reactivity insertions.

Figure 3 compares reactor power for 4.0 mk and 6.7 mk reactivity insertion predicted by the PARET/ANL code in this work and that obtained experiment-

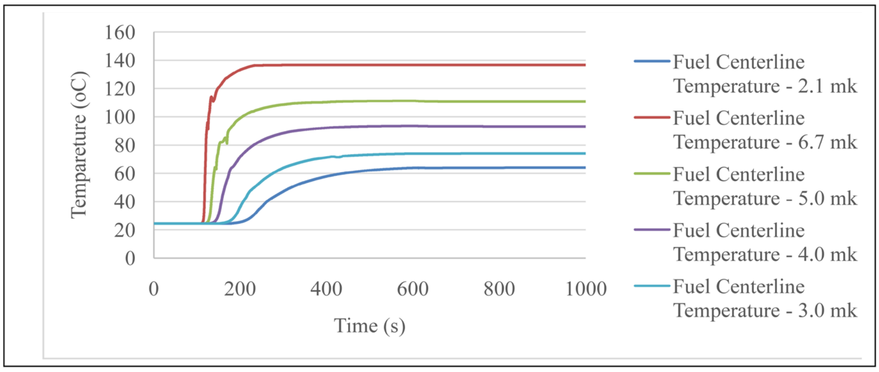

Figure 5. Plot of Fuel center line temperature for 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.71 mk reactivity insertions.

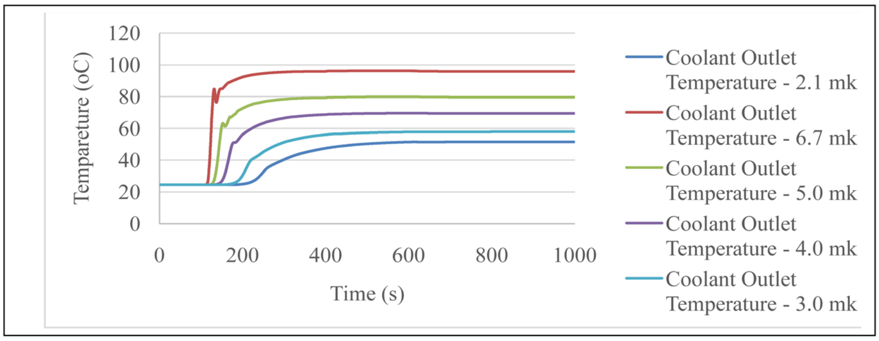

Figure 6. Coolant outlet temperatures for 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.71 mk reactivity insertions.

tally by *Akaho* and *Maaku*. Reactor power predicted under this work for reactivity insertion 6.7 mk has a sharp rise compared to that obtained experimentally. Whereas reactor power for 4.0 mk reactivity insertion has a gradual rise compared to that obtained experimentally. Reactor power for 6.7 mk and 4.0 reactivity insertions predicted in this work compare favorably with that obtained experimentally after 200 s and 400 s, respectively. This time difference could be attributed to the slow rate at which reactor power rises for lower reactivity insertions for both experimental and PARET/ANL code predicted work.

Figure 4 shows a plot of clad temperature for reactivity insertion of 2.1 mk, 3.0 mk, 4.0 mk, 5.0 mk and 6.7 mk. Clad temperature is influenced by reactor power, fuel and coolant temperatures. The noise observed in the plots for reactivity insertions 5.0 mk and 6.7 mk is feedback from reactor power noise observed in Figure 2 for the same reactivity insertion plots.

The graph of fuel centerline temperature is presented in Figure 5. Fuel centerline temperature is influenced by reactor power, clad and coolant temperatures. The noise observed in Figure 2 for reactivity insertion 5.0 mk and 6.7 mk has a feedback effect observed in Figure 5 for the same reactivity insertions. Figure 6 shows the plot of coolant temperature against time and also shows the same trend in noise feedback from reactor power as observed in Figure 4, and Figure 5.

This reactor power noise feedback observed in fuel, clad and coolant temperature plots for reactivity insertions 5.0 mk and 6.7 mk could be attributed to the relationship between component temperature and reactor power also described by Equation (3) for coolant temperature.

5. Conclusions

Simulated transient responses of GHARR-1 after nineteen years of operation indicate good agreement with those reported in the SAR. From the results obtained it can be concluded that the reactor would be safe to operate with reactivity insertion in the range 2.1 mk to 4.0 mk. However, reactor SCRAM settings are such that reactor power should not exceed 120% of the nominal power of 30 kW, and the temperature difference between the core outlet and inlet should not exceed 120% of its nominal limit of 30˚C. Taking into account SCRAM settings, the reactor can be operated with a reactivity insertion of 2.1 mk as the results obtained are within the SCRAM setting limits. The PARET/ANL code is able to correctly predict thermal-hydraulic phenomena below reactivity insertion of 5.0 mk. However, results would be more reliable for longer simulation time for lower reactivity insertions.

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