Steady-State Safety Analysis of Ghana Research Reactor-1 With Low-Enriched-Uranium Core

Edward Shitsi[1]
Graduate School of Nuclear and Allied Sciences,
University of Ghana,
P.O. Box AE1,
Kwabenya, Accra, Ghana;
Nuclear Reactors Research Centre,
National Nuclear Research Institute,
Ghana Atomic Energy Commission,
P.O. Box LG 80,
Legon, Accra, Ghana
e-mail: edwardshitsi@yahoo.com

Prince Amoah
Graduate School of Nuclear and Allied Sciences,
University of Ghana,
P.O. Box AE1,
Kwabenya, Accra, Ghana;
Nuclear Regulatory Authority,
P.O. Box AE 50,
Kwabenya, Accra, Ghana
e-mail: amprince77@gmail.com

Emmanuel Ampomah-Amoako
Graduate School of Nuclear and Allied Sciences,
University of Ghana,
P.O. Box AE1,
Kwabenya, Accra, Ghana;
Nuclear Regulatory Authority,
P.O.Box AE 50,
Kwabenya, Accra, Ghana
e-mail: e.ampomah-amoako@gnra.org.gh

Henry Cecil Odoi
Graduate School of Nuclear and Allied Sciences,
University of Ghana,
P.O. Box AE1,
Kwabenya, Accra, Ghana;
Nuclear Reactors Research Centre,
National Nuclear Research Institute,
Ghana Atomic Energy Commission,
P.O. Box LG 80,
Legon-Accra, Ghana
e-mail: hencilod@gmail.com

Research reactors all over the world are expected to operate within certain safety margins just like pressurized water reactors and boiling water reactors. These safety margins mainly include onset of nucleate boiling ratio (ONBR), departure from nucleate boiling ratio (DNBR), and flow instability ratio (FIR) in addition to the maximum clad or fuel temperature and saturation temperature or boing point of the coolant inside the core of the reactor. This study carried out steady-state safety analysis of the Ghana Research Reactor-1 (GHARR-1) with low enriched uranium (LEU) core. Monte Carlo N-particle (MCNP) code was used to obtain radial and axial power peaking factors used as inputs in the preparation of the input file of plate temperature code of Argonne National Laboratory (PLTEMP/ANL code), which was then used to obtain the mentioned safety parameters of GHARR-1 with LEU core in this study. The data obtained on the ONBR were used to obtain the initiation of nucleate boiling boundary data with respect to the active length of the reactor core for various reactor powers. The obtained results for LEU core were also compared with that of the high enriched uranium (HEU) core. The results obtained show that the 34 kW GHARR-1 with LEU core is safe to operate just as the previous 30 kW HEU core was safe to operate. [DOI: 10.1115/1.4046598]

Keywords: boiling boundary, natural convection, steady-state safety analysis, safety margins, nucleate boiling, heat and mass transfer, thermal systems, two-phase flow and heat transfer

1 Introduction

Plate temperature code of Argonne National Laboratory (PLTEMP/ANL code) has been frequently used to perform the steady-state thermal-hydraulic analysis of research reactors for finding safety parameter such as the peak heat flux, the minimum critical heat flux ratio (CHFR), the minimum flow instability ratio (FIR), and the margin of onset of nucleate boiling (ONBR). The code is intended for thermal-hydraulic analysis of research reactor performance in the subcooled boiling regime. Both turbulent and laminar flow regimes can be modeled [1–4].

The PLTEMP/ANL code, PLate TEMPerature code, was designed and is maintained by the Argonne National Laboratory for thermal-hydraulic studies of research reactors [5]. It was used to calculate the natural circulation flowrate in Ghana Research Reactor-1 (GHARR-1) with high enriched uranium (HEU) fuel [6–8]. Following the core conversion of GHARR-1 from HEU to low enriched uranium (LEU), the miniature neutron source reactor (MNSR) has an increased maximum thermal power level of 34 kW and a peak thermal neutron flux of 1.0×10^{12} n/cm^2 s. The reactor uses rod-type fuel (335 fuel pins, 15 dummies, and 4 tie rods) containing approximately 13% enriched uranium-235 as uranium dioxide (UO_2) pellets in zircaloy-4 cladding. A schematic diagram of the heat transfer and coolant circulation in the reactor is shown in Fig. 1. Cold water enters through the inlet orifice by natural convection. The water flows around the hot fuel elements and comes out via the core outlet orifice. The hot water rises to mix with the bulk of water in the reactor vessel and to the cooling coil. Heat passes through the walls of the reactor vessel to the pool water. Cooling is attained by natural convection using light water. Although the small core has a low critical mass, the presence of a relatively large negative temperature coefficient of reactivity is capable of boosting its inherent safety properties. The small size of the core leads to the leakage and escape of neutrons in both axial and radial directions. These losses are minimized by the beryllium reflectors around the core.

It has become necessary to reduce the amount of high enriched uranium fuels in research nuclear reactors because of the likelihood of these high enriched nuclear materials getting into the wrong hands for nonpeaceful uses and hence the conversion of high enriched uranium fuels to low enriched uranium fuels in research reactors all over the world [9]. Odoi and coworkers [6,7,9] carried out the safety analysis study under both normal and accident conditions for a proposed LEU core of 348 fuel pins under steady-state conditions toward the conversion of HEU core to LEU core of GHARR-1. Reactor safety margins as well as coolant, and clad and fuel temperatures were calculated by setting the inlet temperature to 30 °C. Talha et al. carried out an experimental study and investigated natural convection heat transfer in a plate-type fuel research reactor [10]. It was observed in this study that the

[1]Corresponding author.

Manuscript received September 20, 2019; final manuscript received February 7, 2020; published online March 6, 2020. Assoc. Editor: Cheng-Xian Lin.

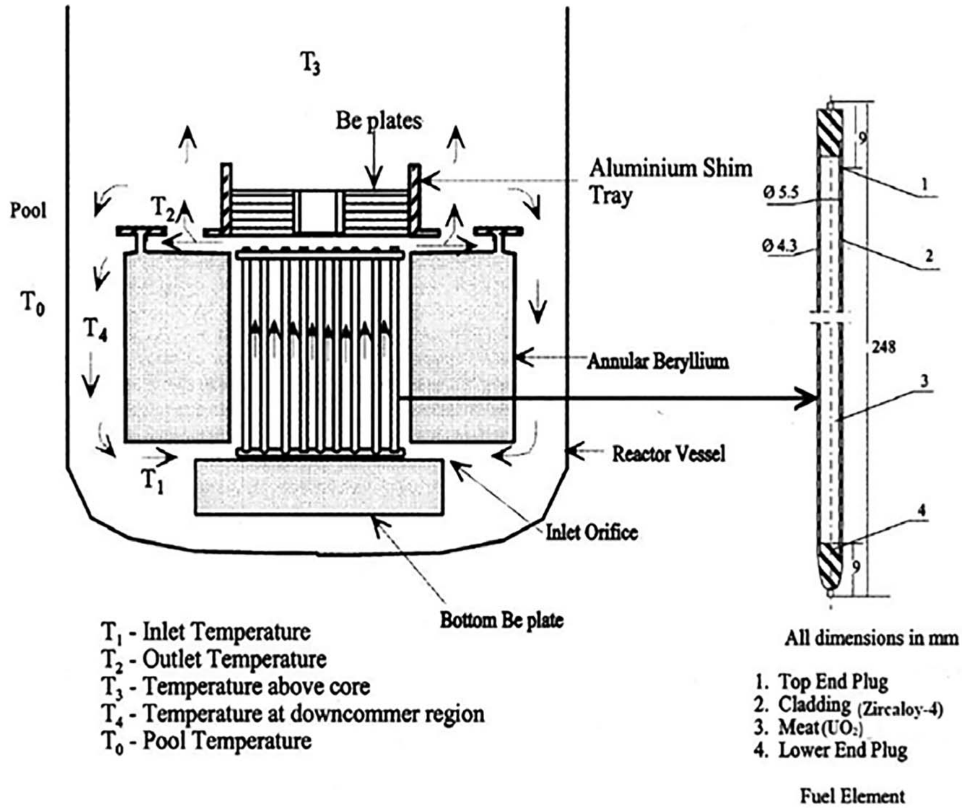

Fig. 1 Heat removal system in GHARR-1 [9]

rate of heat transfer and mass flowrate increase with the heat flux. Mohammed [11] carried out an investigation into thermal-hydraulic parameters of GHARR-1 under steady-state conditions using computational fluid dynamics code. It was observed that the coolant temperature increases with the increasing reactor power.

Although there are few studies in the literature on steady-state safety analysis of research reactors including those of GHARR-1 with HEU core and proposed LEU core with 348 fuel pins, it is necessary to carry out again the steady-state safety analysis of the current state of the GHARR-1 after the conversion of the HEU core with 344 fuel pins and 90.2% U-235 enrichment to LEU core with 335 fuel pins and 13.0% U-235 enrichment. Nucleate boiling boundary analysis with respect to the active core length of the reactor was also included in the present study. The results of the LEU core have been compared with those of the HEU core. Table 1 presents the comparison of key parameters of the LEU core and HEU core. The scope of this work is limited to the steady-state safety analysis, which is applicable to the safety of the operation of research reactors.

2 Methodology

The PLTEMP code is used to compute the steady-state temperature solution for reactors of plate-type fuel separated by coolant channels. The code after various developments is able to obtain a two-dimensional steady-state temperature solution for reactors of various fuel geometries. PLTEMP/ANL incorporates a variety of thermal-hydraulic correlations with which to determine safety margins such as onset of nucleate boiling (ONB), departure from nucleate boiling, and onset of flow instability. Coolant properties for either light or heavy water are obtained from FORmula TRANslation (FORTRAN) functions rather than from tables.

GHARR-I that uses a rod-type fuel can be simulated using this code. PLTEMP is a FORTRAN program that computes the temperature solution for a steady-state flow for a single-fuel assembly or the entire nuclear reactor core. The core is divided into two regions: fueled and nonfueled regions, which are modeled. Coolant channels separate each fuel assembly consisting of one or more plates or rods. A flowchart for the method used in this work is shown in Fig. 2. The PLTEMP input file for GHARR-1 HEU core has been modified to obtain the GHARR-1 LEU input file. The Monte Carlo N-particle (MCNP) was used to obtain radial and axial power peaking factors power peaking factors for 335 fuel pins. The MCNP was also used to obtain maximum power peaking factors for 21 axial segments of the active core of GHARR-1. These power peaking factors were used for the preparation of the PLTEMP input file for the GHARR-1 LEU core.

2.1 Solution of Neutron Particle Transport Equation by Monte Carlo Method.

The neutron particle transport equation is solved by the Monte Carlo Method (MCNP code) to obtain the neutron flux distribution in a reactor core. The neutron flux distribution obtained by MCNP is used for the calculation of power peaking factors. The neutron particle transport equation is a linearized version of the Boltzmann equation [7]. The neutron particle transport equation in a simplified form is based on the neutron conservation equation. The statement of the neutron conservation equation, which relates the net rate of change of the neutron density n with time is given by Eq. (1).

$$\begin{aligned}&\text{Net rate of change per unit volume}\\&= \text{Rate of introduction from sources per unit volume}\\&\quad - \text{Rate of loss by absorption per unit volume}\\&\quad - \text{Rate of loss by leakage per unit volume}\end{aligned} \tag{1}$$

$$\frac{\partial n}{\partial t} = \text{Sources} - \text{Absorption} - \text{Leakage} \tag{2}$$

Table 1 Comparison of key parameters for reference GHARR-1 HEU and LEU cores [9]

Key parameters	HEU	LEU
Reactor type	MNSR	MNSR
Fuel type	Rod	Rod
Power (kW)	30	34
Fuel rod lattice	350	350
Number of active fuel rods	344	335
Number of dummy rods	6	15
Core diameter (mm)	230	230
Core height (mm)	230	230
Fuel lattice pitch (mm)	10.95	10.95
Coolant inlet pressure (atm)	1	1
Coolant heat transfer mode	Natural convection	Natural convection
Reflector	Beryllium	Beryllium
Control rod absorber	Cadmium	Cadmium
Control rod cladding	Stainless steel	Stainless steel
Number of control rods	1	1
Core shape	Cylindrical	Cylindrical
Coolant/moderator	Deionized water	Deionized water
Fuel Meat	$U\text{-}Al_4$	UO_2
U-235 total core loading (g)	~998.1	~1355.3
U-235 enrichment (wt%)	90.2	13.0
U-234 content (wt%)	1.0	0.2
U-236 content (wt%)	0.5	0.25
Density of meat (g/cm^3)	3.456	10.6
Meat diameter (mm)	4.3	4.3
Cladding diameter (mm)	5.5	5.5
Thickness of He gap (mm)	None	0.05
Cladding material	Al-303-1	Zirc-4
Material for grid plates	LT-21	Zirc-4
Top shim tray	LT-21	LT-21
Material for dummy elements	Al-303-1	Zirc-4
Number of tie rods	4	4
Material for tie rods	Al-303-1	Zirc-4
Adjuster guide tubes	4	4
Effective delayed neutron fraction	8.08×10^{-3}	8.57×10^{-3}
Prompt neutron lifetime (s)	8.12×10^{-5}	1.41×10^{-4}
Maximum thermal neutron flux (n/cm^2 s)	1×10^{12}	1×10^{12}
Excess reactivity (mk)	4.0	3.87
Control rod worth (mk)	7.0	6.90
Shutdown margin (mk)	3.0	3.03

$$\frac{\partial n}{\partial t} = S - \Sigma_a\phi - \text{Leakage} \tag{3}$$

$$\frac{\partial n(r,t)}{\partial t} = k_\infty \Sigma_a \phi(r,t) - \Sigma_a\phi(r,t) - \nabla\cdot J(r,t) \tag{4}$$

where $\Sigma_a\phi$ is the absorption, Σ_a is the macroscopic cross section for absorption, k_∞ is the infinite multiplication factor, and J is the rate of loss of neutrons per unit volume.

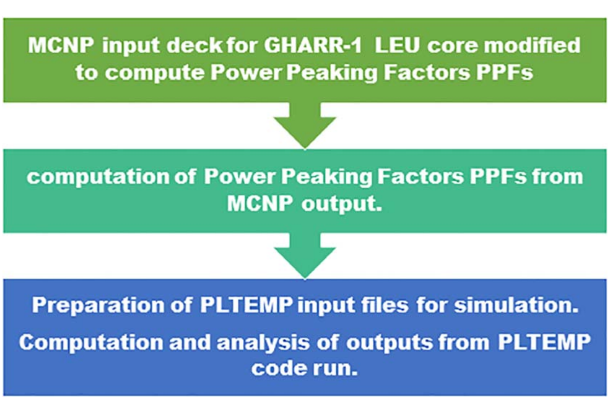

Fig. 2 Flow chart for methodology

Equations (4) is solved to obtain Eqs. (5)–(7), which represent time variation of neutron density or concentration, precursor delayed neutron concentration for six groups of delayed neutrons, and precursor photo neutron concentration for nine groups of photo neutrons, respectively.

$$\frac{dn}{dt} = \frac{(\rho-\beta)}{\Lambda}n_0(t) + \sum_{i=1}^{6}\lambda_i^d C_i^d + \sum_{j=1}^{9}\lambda_j^p C_j^p + S(t) \tag{5}$$

$$\frac{dC_i^d}{dt} = \frac{\gamma^d\beta_i^d}{\Lambda}n(t) - \lambda_i^d C_i^d(t) \tag{6}$$

$$\frac{dC_j^p}{dt} = \frac{\gamma^p\beta_j^p}{\Lambda}n(t) - \lambda_j^p C_j^p(t) \tag{7}$$

where ρ is the net reactivity, n is the net reactor power, β is the effective delayed neutron fraction, γ is the efficiency of delayed and photo neutrons, γ^d and γ^p are the efficiencies of the delayed neutrons and photo neutrons, respectively, β_i^d is the fraction belonging to the ith delayed neutron group, and β_j^p is the fraction belonging to the jth photo neutron group.

2.2 Correlations and Equations Adopted in PLTEMP Code. The relationship involving coolant flowrates through the channels and pressure is given by Eq. (8) [5]:

$$P_2 - P_3 + \frac{W^2}{2\rho_1 A_1^2} - \frac{W^2}{2\rho_3 A_3^2} = g\int \rho_{c,k}(z)\,dz + \frac{K_2 W_{c,k^2}}{2\rho_{c,k}A_{c,k}^2} + \frac{W_{c,k^2}}{2D_{hc,k}A_{c,k}^2}\int\frac{f_{c,k}dz}{\rho_{c,k}(z)} \tag{8}$$

$$\text{for } k = 1,\ 2,\ \ldots N_c$$

$$W = \sum_{k=1}^{N_c} W_{c,k} \tag{9}$$

$$P_2 = P_1 - \frac{W^2}{2\rho_1 A_1^2} - g\rho_1 L_1 - \left(K_1 + \frac{f_1 L_1}{D_{h,1}}\right)\frac{W^2}{2\rho_1 A_1^2} \tag{10}$$

$$P_3 = P_4 - \frac{W^2}{2\rho_3 A_3^2} + g\rho_3 L_3 + \left(K_3 + \frac{f_3 L_3}{D_{h,3}}\right)\frac{W^2}{2\rho_3 A_3^2} \tag{11}$$

where A_F is the flow area of channel (m^2), D_h is the hydraulic diameter based on the wetted perimeter of the channel (m), L is the channel heated length (m), P_1 is the absolute pressure of the creeping coolant in the pool at the assembly inlet level (Pa), P_2 is the absolute coolant pressure just before the inlet to the heated section (Pa), P_3 is the absolute coolant pressure just after the exit from the heated section (Pa), P_4 is the absolute pressure of the creeping coolant in the pool at the assembly exit level (Pa), W is the flowrate in the assembly (total flow in all coolant channels) (kg/s), $W_{c,k}$ is the flowrate in the kth coolant channel (kg/s), and $\rho(z)$ is the coolant density as a function of axial position z (kg/m^3).

The coolant properties and friction factor used in evaluating the integrals in Eq. (8) are evaluated at the temperature $T_{c,k,used}$ (Z) defined by Eq. (12) [5]:

$$T_{c,k,\,used}(Z) = T_{c,k,\,L-1}(Z) + \varepsilon[T_{c,k,\,L}(Z) - T_{c,k,\,L-1}(Z)] \tag{12}$$

where ε is the fraction (e.g., 0.6) of the coolant temperature change from the previous outer iteration.

The temperature relationship involving the temperature on the left side of the fuel meat T_1, the temperature on the right side of the fuel meat T_3, and the maximum temperature of the fuel meat T_m is given by Eq. (13):

$$T_m - T_3 = \{(F_q - 1)s/2k_f\}\{3t_f^2/4 - 2X_{\max}t_f + X_{\max}^2\} \tag{13}$$

where $X_{\max}$ is the location of the maximum fuel temperature, k_f is the fuel thermal conductivity, s is the nominal volumetric heat source strength (W/m^3), t_f is the fuel meat thickness, and F_q is the hot channel factor for heat flux from the meat.

The Collier correlation for mixed convection for calculation of heat transfer coefficient (HTC) is given by Eq. (14). The HTC is used in the calculation of bulk coolant and clad temperatures [5].

$$\mathrm{Nu} = \begin{cases} \mathrm{Nu}_L = \left\{ \max\left[CL1,\ CL2{*}\mathrm{Re}_b^{CL3}\mathrm{Pr}_b^{CL4}\left(\dfrac{\mathrm{Pr}_b}{\mathrm{Pr}_w}\right)^{CL5}\left(\dfrac{g\beta\rho^2 D_e^3(T_w - T_b)}{\mu^2}\right)^{CL6}\right] \right. & \text{if } \mathrm{Re} < RE1 \\ \mathrm{Nu}_T = 0.023\mathrm{Re}_b^{0.8}\mathrm{Pr}_b^{0.4} & \text{if } \mathrm{Re} \geq RE2 \\ \mathrm{Nu}_L + \dfrac{(\mathrm{Re} - RE1)}{(RE2 - RE1)}\{\mathrm{Nu}_T(RE2) - \mathrm{Nu}_L(RE1)\} & \text{if } RE1 \leq \mathrm{Re} < RE2 \end{cases} \tag{14}$$

where the suggested values are as follows: $CL1 = 4.0$, $CL2 = 0.17$, $CL3 = 0.33$, $CL4 = 0.43$, $CL5 = 0.25$, $CL6 = 0.1$, $RE1 = 2000$, and $RE2 = 2500$.

The local value of fuel plate surface temperature or clad temperature T_{surf} is given by Eq. (15) [5]:

$$T_{surf} = T_{in} + \Delta T_b + \Delta T_h \tag{15}$$

where T_{in} is the inlet coolant temperature, ΔT_b is the bulk coolant temperature rise from the inlet of the reactor to the local plate elevation of concern, and ΔT_h is the local temperature rise from the bulk coolant to an immediately adjacent fuel plate surface.

$$\Delta T_b = \frac{q}{WC_p} \tag{16}$$

$$\Delta T_h = \frac{q''}{h} \tag{17}$$

where q is the power added to the coolant from the inlet to the elevation of interest, W is the flowrate in the channel, C_p is the specific heat capacity of the coolant, q'' is the local plate heat flux, and h is the local film coefficient (heat transfer coefficient) at the surface of the fuel plate. Thus, PLTEMP calculates the fuel plate surface temperatures on all fuel plate surfaces at each axial level and compares each temperature to its allowed corresponding value of onset of nucleate boiling temperature T_{onb}.

The Bergles–Rohsenow correlation used for the calculation of ONBR is given by Eq. (18):

$$q_{ONB}(MW/m^2) = 1.0829 \times 10^{-3}P^{1.156}(1.8\Delta T_{sat})^x \tag{18}$$

where P is the coolant absolute pressure (bar), ΔT_{sat} is the wall superheat temperature at ONB, $T_W - T_{sat}$, and $x = 2.16/P^{0.02334}$.

The Mishima–Mirshak–Labuntsov correlation used for the calculation of the CHFR (or departure from nucleate boiling ratio (DNBR)) is given by Eqs. (19) and (20) [5]:

$$q_c = \begin{cases} q_f\left[1 + 2.9 \times 10^5\left(\dfrac{\Delta h_i}{\lambda}\right)^{6.5}\right] \ \text{Mishima fit for } 0 \leq G \leq 200\ \text{kg/m}^2\,\text{s downflow} \\ 0.001A_f\Delta h_i G/A_h\ \text{Mishima fit for } 200 \leq G \leq 600\ \text{kg/m}^2\,\text{s downflow} \\ \text{Min(Mirshak correl, Labuntsov correl) } G \geq 1500\,\text{kg/m}^2\,\text{s downflow} \\ \text{Interpolate between the above if } 600 < G < 1500\ \text{kg/m}^2\,\text{s downflow} \end{cases} \tag{19}$$

$$q_c = \begin{cases} q_f + 0.00146\lambda G\ \text{Mishima fit for one-sided heating, } G \leq 600\,\text{kg/m}^2\,\text{s upflow} \\ q_f + 0.00170\lambda G\ \text{Mishima fit for two-sided heating, } G \leq 600\ \text{kg/m}^2\,\text{s upflow} \\ \text{Min(Mirshak correl, Labuntsov correl) } G \geq 1500\,\text{kg/m}^2\,\text{s upflow} \\ \text{Interpolate between the above if } 600 < G < 1500\ \text{kg/m}^2\,\text{s upflow} \end{cases} \tag{20}$$

$$q_f = \frac{0.7 \times 10^{-3}A_f\lambda(g\rho_v\Delta\rho w)^{1/2}}{A_h[1 + (\rho_v/\rho_l)^{1/4}]^2} \tag{21}$$

where A_f = flow area (m^2), $A_h = P_h L_h$ = heated area (m^2), C_p = specific heat of the coolant (kJ/kg °C), G = mass velocity (kg/m^2 s), g = 9.80665 m/s^2 = acceleration due to gravity (m/s^2), $\Delta h_i = C_p(T_{sat} - T_{in})$ = inlet subcooling (kJ/kg), L_h = heated length (m), P = system pressure (bar), P_h = heated perimeter of the channel (m), q_c = critical heat flux (MW/m^2); q_f = critical heat flux at zero mass velocity (MW/m^2), T_{in} = coolant inlet temperature (°C), T_{sat} = coolant saturation temperature (°C), W = mass flowrate in coolant channel (kg/s), w = width (larger dimension) of the channel rectangular cross section (m), λ = latent heat of vaporization (kJ/kg), ρ_l = saturated liquid density at the system pressure (kg/m^3), ρ_v = saturated vapor density at the system pressure (kg/m^3), and $\Delta\rho = \rho_l - \rho_v$ = density difference between saturated liquid and saturated vapor (kg/m^3).

Babelli-Ishii flow instability criteria of Eqs. (22) and (23) are implemented in the PLTEMP/ANL code for the calcualtion of FIR. More details on the flow instability criteria can be found in the literature of Olson and Kalimullah [5].

$$\frac{N_{sub}}{N_{zu}} = \left(\frac{L_{nvg}}{L}\right)_{critical} + \frac{A_F}{\zeta_H L}\left\{\begin{array}{c} 0.0022\,\mathrm{Pe}\ \text{if } \mathrm{Pe} < 70{,}000 \\ 154\ \text{if } \mathrm{Pe} > 70{,}000 \end{array}\right\} \tag{22}$$

$$\frac{N_{sub}}{N_{zu}} = \left\{ \begin{array}{ll} >1.36 & \text{clearly stable} \\ 1.36 \text{ to } 1.0 & \text{may be stable or unstable} \\ <1.0 & \text{clearly stable or unstable} \end{array} \right\} \quad (23)$$

where N_{sub} = subcooling number, N_{zu} = Zuber number, N_{sub}/N_{zu} = ratio of subcooling number to Zuber number, L = channel heated length (m), L_{nvg} = nonboiling length, i.e., the distance from the start of heated length of channel to the position of net vapor generation (m), L_{nvg}/L = dimensionless nonboiling length, $(L_{nvg}/L)_{critical}$ = critical value of the dimensionless nonboiling length, A_F = flow area of channel (m^2), ζ_H = heated perimeter of channel (m), and Pe = Peclet number.

2.3 Experimental Procedure for GHARR-1. Experimental procedures mainly involve carrying out start procedures and shutdown procedures before and after the reactor operation. The start procedures involve setting all the necessary parameters within the required range before samples are sent into the reactor for irradiation and for neutron activation analysis (NAA). The start procedures ensure that the neutron flux is stable at half power (17 kW) value of 5×10^{11} n/cm^2/s or neutron flux is stable at full power (34 kW) value of 1×1012 n/cm^2/s^1. The coolant temperatures and other parameters are monitored during the operation of the reactor. Figure 3 shows the control console of GHARR-1 with LEU core for reactor control and operation. The shutdown procedures involve following procedures to ensure that the reactor is shut down safely after the reactor operation.

3 Results and Discussion

The GHARR-1 is cooled by natural convection where the coolant water is driven by the density differences of the coolant in the core. The coolant with heavier densities stays at the bottom, whereas the coolant with the lighter densities stays at the top of the core. Thus, the envisaged boiling in the core for large reactor powers beyond the nominal power of 34 kW could only occur from the top of the core because of the natural convection cooling of the core.

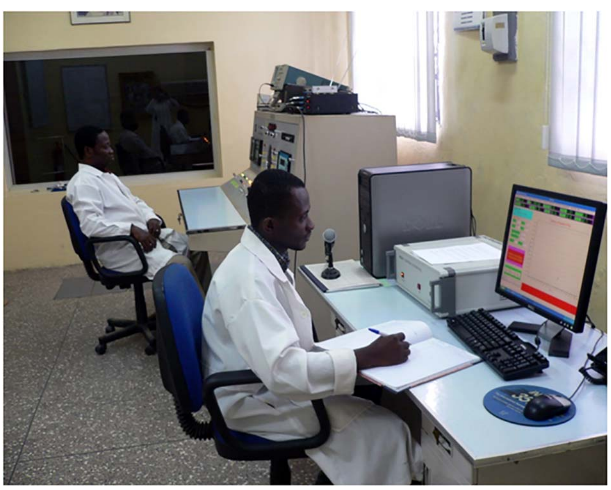

Fig. 3 Control console for reactor control and operation

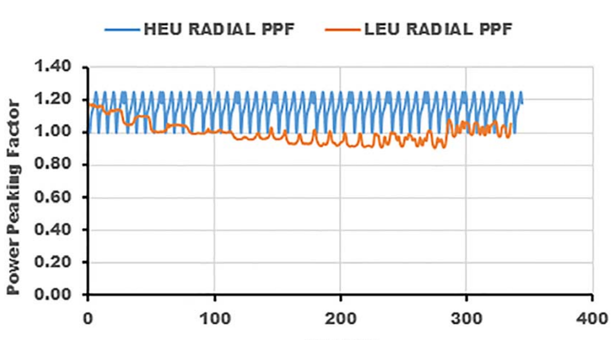

Fig. 4 Radial power peaking factor per pin for 335 fuel pins of the LEU and HEU cores of GHAAR-1

Figure 4 presents the radial power distribution in the LEU core and compared with that of the HEU core. The power profile is nearly flat, indicating core power stability and reduction of neutron leakage as a result of the radial beryllium reflectors. Figure 5 also compares maximum axial power peaking factors for the LEU and HEU cores of GHARR-1. In both cases, the power peaking factors for the LEU core are less than that of the HEU core.

Simulations were run for the nominal LEU core power of 34 kW and other operable powers: 30 kW, 17 kW, and 15 kW. Computed values for the FIR, DNBR, and ONBR are presented in Table 2. It can be observed that the three safety margins, ONBR, DNBR, and FIR, decrease with the increase of the reactor power. Thus, the higher these safety margins, the safer the operation of the reactor at the nominal reactor power. ONBR is the ratio of power at the ONB to the nominal power. DNBR is the ratio of power corresponding to the critical heat flux to the nominal power corresponding to the actual heat flux. FIR is the ratio of power at the onset of flow instability to the nominal power [12–15]. The computed values of the ONBR and the DNBR indicate no possibility of a boiling crisis in the core at the nominal power. The criteria of thermal-hydraulic design for GHARR-1 MNSR require that the DNBR should be greater than 2.55 [15,16]. The values obtained are within the limits of safety for the operating power of the LEU core. The values obtained for the computations of the coolant, clad, and fuel temperatures using 40 °C as the inlet temperature and 1 bar as coolant pressure are also represented in Table 2. These results also indicates good safety margins for the operation of GHARR-1 with the LEU core since the peak temperatures for the clad and the fuel at the nominal reactor power of 34 kW were found to be far below the melting point of the zircaloy-4 clad (1850 °C) and UO_2 fuel (2850 °C).

The geometry adopted for this study is the LEU core of GHARR-1 shown in Fig. 1. In order to validate the accuracy of the numerical results obtained from the usage of PLTEMP code for the steady-state analysis, the coolant temperatures obtained from the code have been compared with that obtained from the

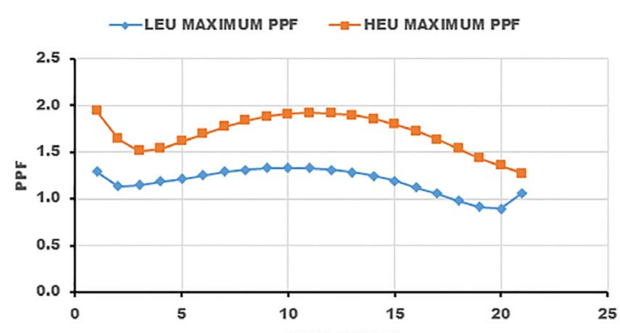

Fig. 5 Maximum power peaking factors of the 21 axial segments for LEU and HEU cores of GHARR-1

Table 2 Steady-state thermal-hydraulic parameters of GHARR-1 with LEU core

Parameter/power	15 kW	17 kW	30 kW	34 kW
Core flowrate (kg/s)	0.317	0.332	0.407	0.425
Coolant temperature (°C)	51.41	52.26	59.33	61.01
Maximum clad temperature (°C)	83.00	86.60	117.91	126.30
Peak fuel temperature (°C)	83.33	86.93	118.24	126.63
ONBR	1.56	1.44	1.27	1.15
FIR	5.53	5.15	3.27	3.00
DNBR	7.65	7.46	5.54	5.34

Table 3 Comparison of outlet coolant temperatures using PLTEMP and LEU experimental data

Inlet coolant temperature (°C)	Outlet coolant temperature (°C), PLTEMP	Outlet coolant temperature (°C), experiment
Power = 17 kW		
38.50	50.73	51.0
40.70	54.15	53.2
41.90	54.68	54.1
42.40	54.53	54.4
43.50	56.57	56.3
Power = 34 kW		
35.30	55.89	54.5
37.40	59.08	56.6
39.10	60.48	57.8
41.50	61.72	59.4
44.50	65.16	62.2

operation of the GHARR-1 with the LEU core. Table 3 presents the comparison for the outlet coolant temperatures of the steady-state simulation using the PLTEMP and experimental data at half nominal power (17 kW) and full nominal power (34 kW). The outlet coolant temperatures are compared well with the experimental values. A deviation in the range of 0.48–4.76% from the experimental values is observed. This indicates a good reliance on the data obtained from the simulation.

In order to have idea about the state of the reactor core at large reactor powers above the nominal power of 34 kW, the safety parameters have been estimated for reactor powers of 40 kW–180 kW, that is up to the reactor power that can cause coolant saturation or boiling of the coolant in the reactor core.

Figure 6 shows the comparison of coolant, clad, and fuel temperatures for LEU and HEU cores of GHARR-1. The coolant, clad, and fuel temperatures increase with the reactor power. These results show that coolant water will begin to boil in the core when the reactor power rises to 180 kW for the LEU core. It was observed that the PLTEMP code could not simulate reactor powers beyond 170 kW for the HEU core. The maximum coolant temperature obtained at 170 kW reactor power for the HEU core was 95.15 °C. This might be due to relatively larger power peaking factors obtained for the HEU core compared to that of the LEU core. The PLTEMP code simulated beyond the 170 kW reactor power for the LEU core. The maximum temperatures for clad and fuel at this saturation condition of 180 kW reactor power are 365.78 °C and 366.11 °C, respectively, for the LEU core. The maximum temperatures for clad and fuel at 95.15 °C with 170 kW reactor power are 327.14 °C and 327.44 °C, respectively, for the HEU core. These clad and fuel temperatures for both the HEU and LEU cores at saturation temperature conditions are relatively far below the melting points of the clad (1850 °C) and fuel (2850 °C). An indication that even at large reactor powers that could cause boiling in the core, the clad and fuel integrity would be maintained and no nuclear incidence or radiation exposure is expected at these large reactor powers. It can also be observed that there is no significant difference in the change in temperature from the fuel meat to the clad for both LEU and HEU cores. The results show that under steady-state and normal operating conditions of the reactor, the fuel meat, clad, and coolant temperatures obtained are slightly higher in the LEU core than that of the HEU core.

Figure 7 shows the comparison of ONBR, DNBR, and FIR for the LEU and HEU cores of GHARR-1. These safety margins decrease with the increasing reactor power. The safety margins FIR and DNBR decrease sharply in the low-power region (the region with no nucleate boiling) and decrease slowly in the large power region (the region with nucleate boiling). The safety margin ONBR decreases slowly in both the low-power and large power regions. The onset of nucleate boiling is expected to begin when the reactor power rises to 43.529 kW for the LEU core and 36.182 kW for the HEU core (ONBR equals one), heat accumulation in the fuel (minimum heat transfer from the fuel pins to the coolant) is expected to begin when the reactor power rises to 73.500 kW for the LEU core and 88.00 kW for the HEU core (DNBR equals one), and flow instability will not occur in the core for both the LEU and the HEU cores (FIR more than one for the reactor power that can cause boiling in the core).

Figure 8 shows the comparison of nucleate boiling boundary results for the LEU core and the HEU core of GHARR-1. It was observed from Fig. 8 that nucleate boiling boundary decreases with the increasing reactor power. The higher the nucleate boiling boundary, the safer the operation of the reactor. These results show that boiling will not occur in the core when the reactor power is below 43.529 kW for the LEU core and 36.182 kW for the HEU core (nucleate boiling boundary equals 230 mm). At the point of coolant water saturation in the core, the reactor power will rise to 180 kW with te nucleate boiling boundary of 50.14 mm for the LEU core. Thus, nucleate boiling initiation is expected to occur at 50.14 mm from the bottom of the core and full boiling is also expected to occur at the top of the core at this 180-kW reactor power for the LEU core. At the maximum coolant temperature of 95.15 °C, the PLTEMP code could simulate for the HEU core, the reactor power is 170 kW with nucleate boiling boundary of 56.35 mm. Nucleate boiling initiation is expected to occur at 56.35 mm from the bottom of the core, but full boiling is not expected to occur at the top of the core at 170 kW for the HEU core.

The steady-state safety analysis is very useful and applicable to the operation of research reactors in order to determine the

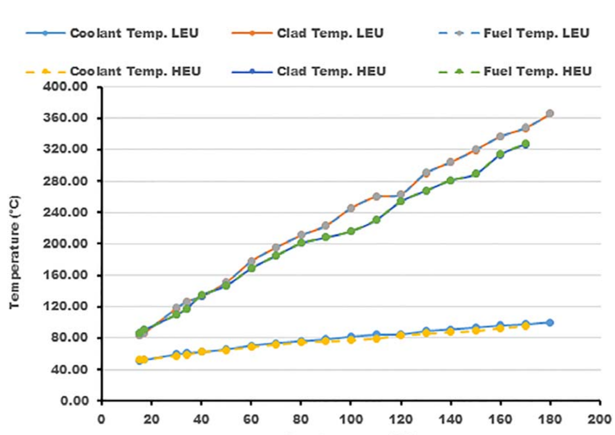

Fig. 6 Comparison of coolant, clad, and fuel temperatures for LEU and HEU cores

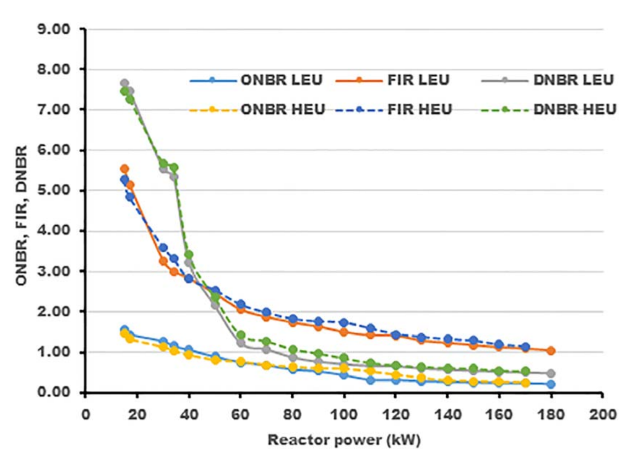

Fig. 7 Comparison of ONBR, DNBR, and FIR results for LEU and HEU cores

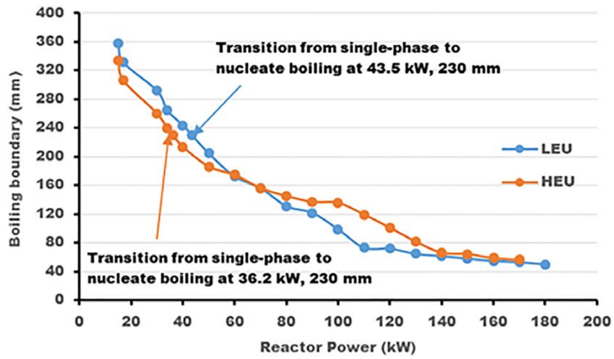

Fig. 8 Boiling boundary results for LEU and HEU cores of GHARR-1

margins of safety for the operation of these research reactors at particular reactor power. The larger the difference between one and the values of these safety margins, the safer the operation of these research reactors at particular reactor power. In addition, the ONBR could be used to determine the boiling boundary showing the regions of the single-phase flow region and nucleate boiling region at particular reactor power. Research reactors are normally designed to operate fully in a single-phase flow region. For research reactors, the margin to nucleate boiling tends to be the most limiting criterion [5].

4 Conclusion

The steady-state safety analysis of GHARR-1 with LEU core was carried out in this study. The results of the LEU core have been compared with that of the HEU core. The MCNP/ANL and PLTEMP/ANL codes were used to obtain the results for this safety analysis. The results show that the LEU core is safe to operate just like the HEU core. Good safety margins exist for the operation of the LEU core at a nominal power of 34 kW (ONBR equals 1.15, DNBR equals 5.34, and FIR equals 3.00). These safety margins decrease with the increasing reactor power, an indication that the larger the safety margin, the safer the operation of the reactor. Boiling is expected to occur at top of the LEU core when the reactor power rise to 180 kW with the nucleate boiling boundary of 50.14 mm from the bottom of the core. The maximum temperatures for clad and fuel at 180 kW are 365.78 °C and 366.11 °C, respectively, for the LEU core. The PLTEMP code could not simulate up to the boiling point or saturation temperature condition for the HEU core. The maximum temperatures for clad and fuel at 170 kW with the corresponding 95.15 °C coolant temperature are 327.14 °C and 327.44 °C, respectively, for the HEU core. At the maximum coolant temperature of 95.15 °C, the nucleate boiling boundary is 56.35 mm from the bottom of the core. The results also indicate good safety margins for the operation of GHARR 1 with the LEU core since the peak temperatures for the clad and the fuel were found to be far below the melting point of the zircaloy −4 clad (1850 °C) and UO_2 fuel (2850 °C) even at large reactor powers.

Acknowledgment

The Los Alamos National Laboratory (LANL) and ANL are acknowledged for making available MCNP5 and PLTEMP codes used for carrying out this study. Nuclear Reactors Research Centre NRRC of Ghana Atomic Energy Commission GAEC is also acknowledged for making available the steady-state experimental results used for comparison of the numerical results obtained in this work.

References

[1] Springfels, D., Jordan, K. A., and Schubring, D., 2015, "Comparison of Thermal Hydraulic Analysis Methods (CODES) for the University of Florida Training Reactor (UFTR)," International Topical Meeting on Nuclear Reactor Thermal-Hydraulics NURETH-16, Chicago, IL, Aug. 30–Sept. 4, pp. 1–12.

[2] Kadalev, S., 2012, "Thermal Hydraulic Calculations of the IRT-Sofia Research Reactor with LEU Fuel IRT-4M, Sofia," C. R. Acad. Bulg. Sci., **65**(9), pp. 1185–1192.

[3] Rahman, M. M., Akond, M. A. R., Basher, M. K., and Huda, Q., 2014, "Steady-State Thermal-Hydraulic Analysis of TRIGA Research Reactor," World J. Nucl. Sci. Technol., **4**(2), pp. 81–87.

[4] Costa, A. L., De Lima Reis, P. A., Maria, C. P., Veloso, A. F., and Da Silva, C. A. M., 2014, "Research Reactor Analysis Using Thermal Hydraulic and Neutron Kinetic Coupling," International Conference on Nuclear Engineering, Vol. 2A1, Prague, Czech Republic, July 7–11, pp. 1–5.

[5] Olson, A. P., and Kalimullah, M., 2010, "A Users Guide to the PLTEMP/ANL V4.0 CODE, Global Threat Reduction Initiative (GTRI)—Conversion Program," Nuclear Engineering, Division, Argonne National Laboratory, Argonne, IL, pp. 1–214.

[6] Odoi, H. C., Akaho, E. H. K., Nyarko, B. J. B., and Abrefah, R. G., 2014, "Current Studies of Conversion of Ghana Research Reactor Core to Low Enriched Uranium," Ghana Atomic Energy Commission, Ghana, Technical Report, pp. 1–8.

[7] Odoi, H. C., 2014, "Reactor Core Conversion Studies of Ghana Research Reactor—1 and Proposal for Addition of Safety Rod," PhD dissertation, University of Ghana, Ghana, pp. 1–174.

[8] Mweetwa, B. M., 2015, "Evaluation of Safety Parameters for Gharr-1 After Nineteen (19) Years of Operation Using REBUS/ANL, MCNP5, PARET/ANL and PLTEMP/ANL Simulation Codes," MPhil thesis, Graduate School, University of Ghana, Ghana, pp. 1–157.

[9] Odoi, H. C., Akaho, E. H. K., Jonah, S. A., Abrefah, R. G., and Ibrahim, V. Y., 2014, "Study of Criticality Safety and Neutronic Performance for a 348-Fuel-Pin Ghana Research Reactor-1 LEU Core Using MCNP Code," World J. Nucl. Sci. Technol., **4**(1), pp. 46–52.

[10] Talha, K. E. A., Abdel-Hadi, E. A., El-Ghnam, R. I., Taher, S. H., and Abdel-Latif, S. H., 2012, "Thermal Hydraulic Behavior of Research Reactor During Natural Convection Cooling Mode," Arab J. Nuclear Sci. Appl., **45**(4), pp. 243–256.

[11] Mohammed, S., 2015, "Investigation of Heat Transfer and Distribution in the Core of Ghana Research Reactor-1 (GHARR-1) Using STAR-CCM+ CFD Code," M.Phil. thesis, Graduate School, University of Ghana, Ghana, pp. 1–82.

[12] Omar, H., Ghazi, N., and Hainoun, A., 2012, "Thermal Hydraulic and Safety Analysis for Core Conversion (HEU-LEU) of Syrian Miniature Neutron Source Reactor," Prog. Nucl. Energy, **60**, pp. 140–145.

[13] Hainoun, A., Doval, A., Umbehaun, P., Chatzidakis, S., Ghazi, N., Park, S., Mladin, M., and Shokr, A., 2014, "International Benchmark Study of Advanced Thermal Hydraulic Safety Analysis Codes Against Measurements on IEA-R1 Research Reactor," Nucl. Eng. Des., **280**, pp. 233–250.

[14] IAEA, 2016, "Safety of Research Reactors," Specific Safety Requirements No. SSR-3, pp. 1–152.

[15] Shitsi, E., Akaho, E. H. K., and Nyarko, B. J. B., 2018, "Theoretical Investigations of Two-Phase Flow and Heat Transfer in Parallel Multichannel Core of a Low Power Reactor," Issues Sci. Res., **2**(1), pp. 1–17.

[16] Akaho, E. H. K., Anim-Sampong, S., Dodoo-Amoo, D. N. A., Maakuu, B. T., Emi-Reynolds, G., Osae, E. K., Boadu, H. O., and Bamford, S. A., 2003, "Ghana Research Reactor-1 Final Safety Analysis Report," Ghana Atomic Energy Commission, Technical Report, GAEC-NNRI, Vol. RT-90, pp. 1–353.