Current Studies of Conversion of Ghana Research Reactor Core to Low Enriched Uranium

H. C. Odoi, E. H. K. Akaho, B. J. B. Nyarko, R. G. Abrefah, E. Ampomah-Amoako, R. B. M. Sogbadji, S. A. Birikorang, J. Gbadago.
National Nuclear Research Institute
Ghana Atomic Energy Commission
Kwabenya, Accra – Ghana.
and
J. E. Matos, J. Liaw, J. Morman, M. Kalimullah, S. C. Mo.
Argonne National Laboratory
Argonne, IL 60439 USA

ABSTRACT

The core conversion studies of the Ghana Research Reactor Core started in 2006 and with the objective of designing an LEU core with similar operational capabilities as the original HEU core and with acceptable safety margins under both normal and accident conditions. Steady-state studies for converting the Ghana MNSR facility from HEU to LEU core have been performed using the PLTEMP/ANL V4.1 code. Six hot channel factors are used in the code to calculate reactor safety margins as well as clad surface and coolant temperatures using an inlet temperature of 30 °C. Other results of thermal-hydraulics analyses for both HEU and LEU cores, such as the minimum onset of nucleate boiling ration and critical heat flux, are shown in this report.

1. Introduction

The core conversion studies of the Ghana Research Reactor Core started in 2006 and with the objective of designing an LEU core with similar operational capabilities as the original HEU core and with acceptable safety margins under both normal and accident conditions. The thermal-hydraulic design of the reactor is closely related to the neutronic calculations and the structural design [1]. The geometrical dimensions of the fuel element, fuel element temperatures, coolant pressure, temperatures and velocities are analytically shortened to satisfy the requirements of reactor safety during all operational states.

The core region of GHARR-1 is located 4.7 m under water close to the bottom of a watertight reactor vessel. The quantity of water is 1.5 m^3 in the vessel, which serves the purpose of radiation shielding, moderation and as primary heat transfer medium. In addition, heat can be extracted from the water in the vessel by means of a water-cooling coil located near the top of the vessel. The water-filled reactor vessel is in turn immersed in a water-filled pool of 30 m^3. Cold water is drawn through the inlet orifice by natural convection. The water flows past the hot fuel elements and comes out through the core outlet orifice. The hot water rises to mix with the large volume of water in the reactor vessel and to the cooling coil. Heat passes through the walls of the container to the pool water. A diagrammatic representation of the heat transfer mechanism is represented in Fig. 1.

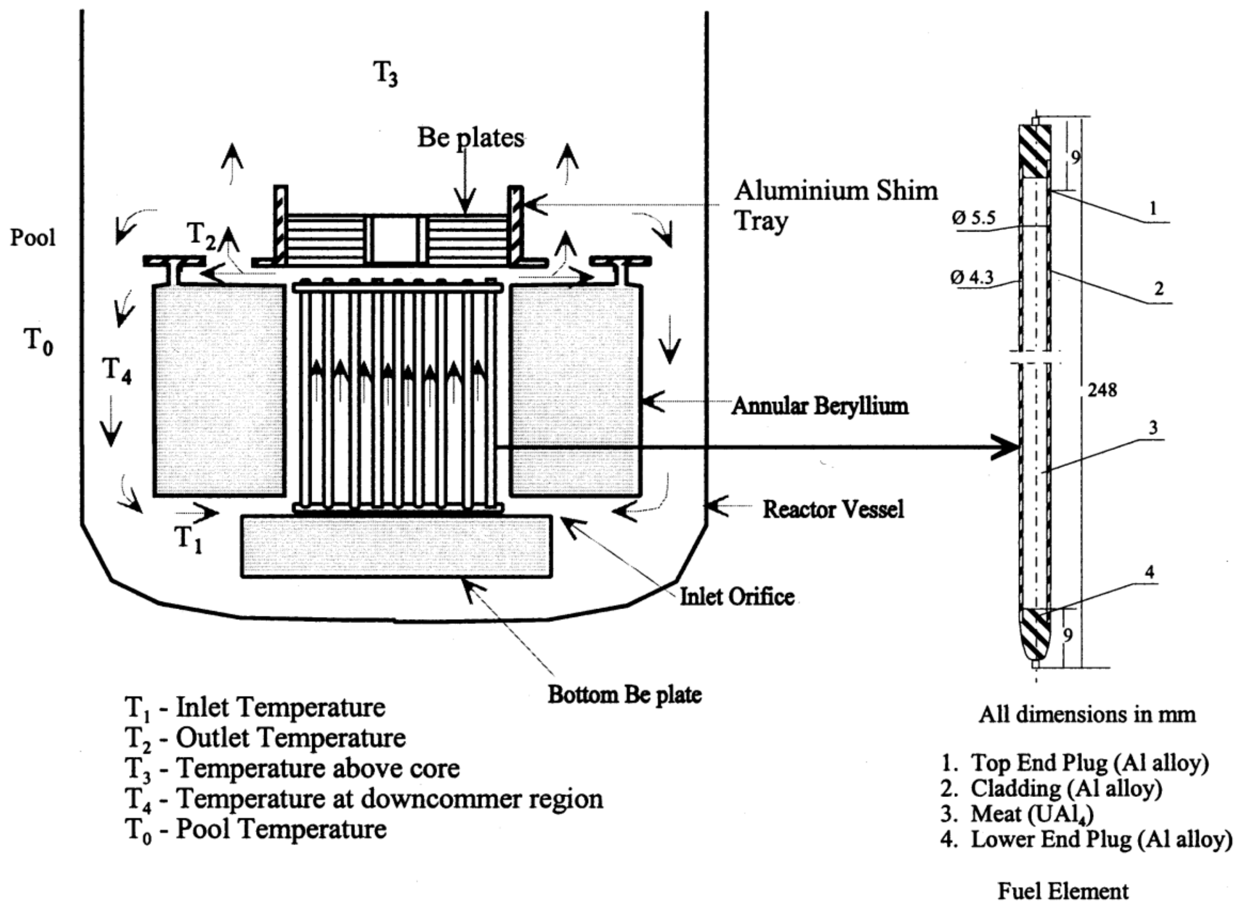

Fig.1. A schematic diagram of the coolant flow pattern [2].

The core inlet flow orifice impedes the natural circulation of water through the core. Its area is fixed during assembly of the reactor and it is deliberately chosen such that the highest power achieved during the design basis self-limiting power excursion can cause no damage to the core or present any hazard to staff about the reactor.

The GHARR-1 reactor has a small core. It can be shown that the coolant flow in the core is at the transient phase from laminar flow to turbulent flow. The flow transition will occur when there is an increase in power. The closer to the upper part of elements, the stronger the turbulence becomes. The buoyancy force in natural circulation must overcome the friction. Calculations show that the friction resistance is small, about 10% of the total resistance. Meanwhile, the inlet resistance is about 70% of the resistance and thus has a great effect on the state of flow. An appropriate choice for the inlet flow orifice is a very important factor in the thermal-hydraulic design. As inlet orifice gets smaller, the flow velocity and in turn, the turbulence in the core will increase. As a result, the heat transfer from fuel element to coolant will be improved. However, a smaller inlet orifice will cause an increase in resistance and a decrease in flow rate resulting in a rapid increase of temperature. An appropriate temperate rise is an essential condition. Furthermore, a too high temperature rise will cause a relatively large temperature effect and make the excess reactivity used for compensating xenon poisoning too small and thus the operable time will be reduced. The relationship obtained using the thermal-hydraulic test data is expressed in the form [1]

$$\Delta T = 6.81 P^{(0.59+0.0019 T_i)} T_i^{-0.35} \tag{1}$$

where P is the reactor power in kW, T_i is the coolant temperature (°C) and ΔT is the temperature difference between the inlet and outlet coolant in the core.

It is evident from this relationship that the increase in the temperature difference between the inlet and outlet coolant ΔT will increase with increasing reactor power and decrease with the inlet coolant temperature.

The water in the reactor is not pressurized and it relies upon natural convection. The problems of de-pressurization or coolant flow pump failure are not posed. A water-cooled coil of limited heat removal capacity removes the thermal energy generated in the core eventually. Due to the fact that the reactor possesses limited excess reactivity and reactivity feedback characteristics, any significant deterioration in heat removal capability will eventually result in an automatic decrease in reactor power to match up with the new heat removal capacity.

The temperature in an operating reactor varies from point to point within the system. As a consequence, there is always one fuel rod (usually one near the centre of the reactor) that at some point along its length is hotter than all the rest. This maximum fuel temperature is determined by the power level of the reactor, the design of the coolant system, and the nature of the fuel. [3]. One major design of a reactor coolant system is to provide for the removal of heat produced at the desired power level while ensuring that the maximum fuel temperature is always below this predetermined value. This in tend ensures good safety margin. Under subcooled flow boiling conditions, the boiling crisis is often called departure from nucleate boiling (DNB). The heat flux at which the boiling crisis occurs is named the critical heat flux (CHF). In general, thermal performance improvements are highly desirable and it is therefore needed to predict CHF accurately at the earliest stages of a new product design. In the case of a nuclear reactor core, CHF margin gain (e.g. using improved fuel assembly design) can allow power uprate and enhanced operating flexibility [4]. Most metal-finishing operation score tiny grooves on the surface, but they also typically involve some chattering or bouncing action, which hammers small holes into the surface. When a surface is wetted, liquid is prevented by surface tension from entering these holes, so small gas or vapour pockets are formed. These little pockets are sites at which bubble nucleation occurs [5]. The ONB is not a limiting criterion in the design of a fuel element. However, it is a heat transfer regime which should be identified for proper hydraulic and heat transfer considerations, i.e., single-phase flow versus two-phase flow. For reactor design purposes, acceptable data on burnout heat flux are needed since departure from nucleate boiling (DNB) is potentially a limiting design constraint. Optimization of core cooling against other neutronic, economic, and materials constraints can best be accomplished by judicious use of standard, experimentally-deduced DNB correlations [6]. The parameter most used to evaluate margin to failure by boiling crisis is the critical heat flux ratio (CHFR), or departure from nucleate boiling ratio (DNBR). This is the ratio of critical heat flux calculated by the correlation to the most limiting heat flux condition in the reactor [7].

2. Method of Analysis

Four input data files were used in the PLTEMP/ANL V4.1 code to calculate the safety margins in the steady-state operation of GHARR-1 with HEU core. In addition, an input file giving the axial power shape of the fuel pin modeled (the average power pin or the peak power pin in the HEU core) was also used with the four input data files. Another set of four similar input data files were used to calculate steady-state safety margins of GHARR-1 with LEU core at both 30 kW and 34 kW. In addition, an input file giving the axial power shape of the fuel pin modeled (the average power pin or the peak power pin in the LEU core) was also used with each set of the four input data files; this is required by the PLTEMP/ANL V4.1.

One set of input models one (average fuel pin) of the 344 or 348 fuel pins in the HEU or LEU core respectively with a reactor power of 15 kW and a coolant inlet temperature of 24.5 °C. The pin is modeled as a solid rod of radius 2.15 mm in a 0.6 mm thick cladding, without any gap resistance in the case of HEU core. This input data file was used to calibrate the hydraulic resistance in the PLTEMP/ANL model to reproduce an experimentally measured coolant temperature rise of 13 °C (from 24.5 °C to 37.5 °C).
Another input data file uses the above determined value of the hydraulic resistance coefficient, and models one (average fuel pin) of the pins in the HEU and LEU cores respectively when the reactor is operating at the nominal reactor power of 30 kW. The purpose of this input data file is to raise and adjust the coolant channel inlet temperature so that the coolant exit temperature is 70 °C. The next input data file uses the above adjusted values of the hydraulic resistance coefficient and the channel inlet temperature, and models the peak power pin of the core, with six hot channel factors (HCF). The purpose of this input data file is to determine the maximum allowed operating reactor power with all hot channel factors applied.

The final set of input data files is identical to the third set of input data files except that five of the hot channel factors have been set to 1.0 in order to calculate the maximum allowed reactor power without hot channel factors. The hot channel factor for power was kept unchanged at its actual value because the ratio of the peak pin to the average pin power is certain. Using this input data file, the pin power was raised and adjusted so that the minimum ONBR on the cladding outer surface is exactly 1.0. The minimum ONBR occurs in axial node 10. When this minimum ONBR is 1.0, the pin power multiplied by number of pins gives the maximum allowed operating reactor power of the core without hot channel factors.

Six hot channel factors (defined below) are used in the PLTEMP/ANL V4.1 code to calculate research reactor safety margins. These factors are different in natural circulation flow from those in forced flow. The basic reason for this is that in natural circulation the coolant flow is induced by the power produced in the pin (thus softening the effect of pin power on inlet-to-outlet coolant temperature rise) whereas it is not so in forced flow. In forced flow, the pressure drop induces the coolant flow. The hot channel factors for forced flow over research reactor fuel plates have been formulated earlier [8, 9]. Table 1 shows the type of uncertainties included in each of the six hot channel factors. The uncertainties of pool water level and pin heated length are not included.

Table 1. Uncertainties Included in the Six Hot Channel Factors

No.*	Uncertainty Type	FPOWER	FFLOW	FNUSLT	FBULK	FFILM	FFLUX
1	Neutronics calculation of power density in a pin, u_1				X	X	X
2	U-235 mass per pin, u_2				X	X	X
3	UO_2 pellet radius, u_3					X	X
4	U enrichment in a pellet, u_{10}					X	X
5	UO_2 pellet density, u_{11}						
6	Fuel pin radius, u_{12}				X	X	X
7	Fuel pin pitch, u_{13}				X	X	
8	Flow redistribution among channels, u_6				X		
9	Reactor power measurement uncertainty, u_7	X					
10	Flow uncertainty due to uncertainty in friction factor, u_8		X				
11	Heat transfer coefficient uncertainty due to uncertainty in Nu number correlation, u_9			X			

*1 – 8 are for local or random uncertainties whiles 9 – 11 represent system-wide uncertainties

System-wide or Global Hot Channel Factors:
FFLOW = a factor to account for the uncertainty in total reactor flow
FPOWER= a factor to account for the uncertainty in total reactor power
FNUSLT = a factor to account for the uncertainty in Nusselt number correlation

Local Hot Channel Factors:
FBULK= a hot channel factor for local bulk coolant temperature rise
FFILM = a hot channel factor for local temperature rise across the coolant film
FFLUX = a hot channel factor for local heat flux from cladding surface

3. Results and discussion

The reactor power at minimum onset of nucleate boiling ratio, ONBR=1 without hot channel factors is 65.72 kW for the HEU and 67.75 kW for the LEU, further more the reactor power at ONBR=1 with all six hot channel factors are 51.6 kW and 53 kW for the HEU and LEU core respectively. The minimum departure from nucleate boiling ratio with all six hot channel factors is 8.9 for the HEU and 8.5 for the LEU core.

The onset of nucleate boiling ratio (ONBR) and departure from nucleate boiling ratio (DNBR) computed so far indicate there is no boiling in both cores. And these indicate the limits of operating power for both the HEU and LEU cores. This is the *true* allowed power in the sense that there is no allowance in it for any error in the power measuring instrument. This maximum allowed operating power assumes that the power measuring instrument is perfect without any measuring error. The results also indicated good safety margins so far as the boiling point of the coolant and the melting points of both the fuel and clad are concerned.

Thermal hydraulic parameters obtained from further studies undertaken on both the HEU and LEU cores at nominal reactor powers are show in table 2. The results of the calculations for the clad surface and coolant temperatures using an inlet temperature of 30 °C and a coolant pressure of 1 bar are also shown in this table.

Table 2. Comparison of HEU and LEU steady-state parameters using PLTEMP/ANL

Parameter	**HEU – 344 rods**	**LEU – 348 rods**	**LEU – 348 rods**
Power (Kw)	30.0	30.0	34.0
Core Flow Rate (Kg/S)	1.1E-3	1.1E-3	1.2E-3
ONBR Minimum	1.696	1.237	1.190
DNBR Minimum	14.56	14.306	14.352
Max. Clad Surface Temp. (°C)	77.3	95.0	98.3
Max. Coolant Temp. (°C)	53.1	53.4	57.1

For the LEU core the nominal power is raised to 34 kW in order to meet the flux level of 1×10^{12} n/cm2.s. Hence the computations, using PLTEMP, were done for the LEU core at this power and the steady-state parameters were also compared with those of HEU and LEU at 30 kW (table 2).

Thermocouples and level gauges are used for measuring the reactor thermo-hydraulic parameters. These devices can monitor the operating conditions of the reactor and provide information through the instruments mounted on the main control console. Measurement of the temperature difference between core inlet and outlet is accomplished with 2 Alumel-Chromel thermocouples. The coolant outlet temperatures at various reactor powers and coolant inlet temperatures are compared in table 3; these are computations were made using the PLTEMP.

Table 3. Comparison of computed coolant outlet temperatures at various powers and inlet temperatures

Power (kW)	**Inlet Temp. (ºC)**	**HEU Outlet Temp. (ºC)**	**LEU Outlet Temp.(ºC)**
0.3	32.0	33.2	33.2
	35.0	36.1	36.1
3	32.0	37.0	37.0
	37.0	41.7	41.7
	39.0	43.6	43.6
15	30.0	43.9	44.0
	37.0	50.0	50.2
	42.0	54.6	54.7
30	30.0	51.4	51.7
	34.0	54.7	55.4

37.0	57.3	57.5
42.0	61.6	61.9

The safety settings of the reactor ensure that protective action will correct an abnormal situation before a safety limit is exceeded [1]. For the HEU, the safety system settings for reactor thermal power, P, height of water above the top of the core, H, and ΔT are as follows:

P(max) = 36 kW
H (min) = 465 cm
ΔT(max) = 21 ºC

The effect of inlet temperature on temperature difference, as computed by PLTEMP, for both HEU and LEU are shown in table 4.

Table 4. Effect of Inlet Temperature on Temperature Difference at Nominal Operating Power for the HEU and LEU Cores

T_{IN} (ºC)	30 kW		36 kW	
	HEU – ΔT (ºC)	LEU – ΔT (ºC)	HEU – ΔT (ºC)	LEU – ΔT (ºC)
10	29.02	29.15	27.00	32.28
15	27.03	27.16	24.20	30.20
20	25.47	25.59	22.66	28.54
30	23.14	23.26	20.97	26.03
35	22.25	22.37	20.63	25.07
40	21.49	21.61	20.54	24.24

4. Conclusion

The PLTEMP/ANL V4.1 code is used to calculate the safety margins in the steady-state operation of GHARR-1 for both HEU and LEU cores. Results show that steady state analysis for the two cores are comparable. Thermal hydraulics analysis indicates good safety margin for the LEU core. Neutronic analysis has been dealt with in other platform.

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