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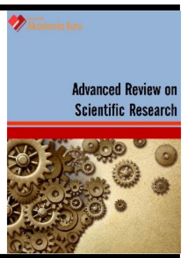

Control rod safety management in nuclear power plant: A review

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ABSTRACT

The primary objective of control rod management is to ensure the safe, reliable and optimum use of the nuclear fuel in the reactor, while remain within the limits imposed by the design of the fuel assembly and reactor w.r.t the safety analysis. In numerous reactors, the control rods perform the function of reactivity control, both globally and locally, latter also control the power distribution of the core. Most control rods are completely withdrawn from the core during operations and fully inserted during shutdown of the reactor. This are the prove of an investigative study into optimization of the heterogeneous control rods, the safety management of an additional safety rod, and control rod drop hydrodynamic analysis which will be able to provide an efficient and maximum safety procedure for emergency shut down system in the reactor. It is also will be very important safety features of the reactor. To design the additional safety rod at the inner irradiation sites with boron carbide (absorber material) and stainless steel (clad) and also to design a simple heterogeneous control rod design to analyse the effects of geometry self-shielding, the Monte Carlo Neutron Particle Code, version 5 (MCNP5) was used. By simulating the unsteady flow field around the control rod, the hydrodynamic analysis of the control rod can be analysed. A correlation based on the achieved data was proposed to provide useful information on the safety management during the research.

Keywords:
Nuclear power plant, safety management, core, control rod

1. Introduction

Nowadays, there are about 447 reactors that operating worldwide [1]. There are various criteria used for classification, which mainly based on neutron spectra, types of fuel, and types of coolant. Along with the times, the new generation of various types of reactors are developed for many purposes, such as for electricity generation, research and development, and also for educational purpose. In spite of all arising technologies for a better performance in energy production, safety aspect will always be questioned and become the most important issue to be concerned with.

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One of the important safety analysis that involves the most critical part in nuclear reactor is the reactivity control analysis. Reactivity is a parameter used to indicate the departure from criticality of the core. Reactivity control is important in order to prevent reactivity initiated accident from occurs, and to mitigate them if it occur. The occurrence of reactivity initiated accident may causes significant increase in reactivity which produces enhancement in temperature and power which will lead to the damage of the core [2]. Thus, the reactivity control is needed in order to maintain negative reactivity of the core to ensure safety operation.

2. Control Rod Safety Management

2.1 Control rod drop in Pressurized Water Reactor (PWR)

The control rod drop analysis is a crucial assessment for safety analysis due to the possibility of the rod assemblies to accidently to fall within specified time and displacement, from a fully withdrawn condition inside the core especially during reactor is operating. A set of accident is generated. The precise location and motion history of control rod are the factors that important for a proper investigation of flux of neutron in the core during the occurrence of the accident. All of the related informations were determined by using hydrodynamic analysis using the guide tube. Under the generated accident conditions, the interaction between the control rod and the fluid in contact to it with a small different in distance between the control rod and the guide tube, would be of very helpful to obtain these informations in order to prevent any postulated accidents related to sudden change of reactivity. Based on the study made by Ataollah Rabiee *et al*. [2], the objective is to evaluate what will happen when a control rod falls into the reactor core during the accident hydro-dynamically.

As proposed by Groudev and Stefanova [3], an analysis was made by using RELAP code to simulate sudden drop of control rod assembly to its fully inserted position. The results obtained are respected to the VVER 440 (Unit #2 at Kozloduy site), in which RELAP 5 model were used in comparing the outcomes with the experimental transient data. The comparison between the results shows a good agreement [4].

Yoon *et al*. [5] proposed to use fluid structure interaction as a new procedure to analyse the occurrence of control rod to accidentally drop for PWR power plant. Fluid structure interaction boundary conditions are applied because of the submerged control rod in a guide tube of a fuel assembly. Their analysis gives useful results for the design of the control rod in the fuel assembly. Some of the works can be mentioned in the references such as Andersson *et al*. [6], Kurihara [7], Zhang *et al*. [8].

Andriambololona *et al*. [9] proposed a new methodology to simulate insertion or drop of the cluster control rod assembly into its guides. The outcome of this analysis shows a good agreement of the numerical results compared to experimental measurements. There are many loads were taken into account, such as fluid load gravity and also the frictional force between the control rod and the guide.

Fritz [10] comes with a methodology known as POLCA 7, that is used to simulate a set of accidents due to control rod drop which specifically developed for Forsmark Nuclear Power Plant in cooperation with Westinghouse. This work aimed to investigate what will happen if a control rod drops during hot zero power occur in a reactor.

Souza *et al*. [11] presented the recognizing patterns in the neutron ex-core responses of detector which allows online identification if control rod drops into the core when the reactor is operating. The outcomes of the study as well as the detector’s behaviour and responses were demonstrated and used to support the feasibility of this method.

Eissa *et al*. [12] did an investigation in order to obtain the optimal positions for detectors used for monitoring purposes, which is by depending on the calculated correlation coefficient for the flux of thermal neutron in case if each control rod group is fully inserted or accidentally dropped into the core while the reactor is operating at full power, relative to the flux of thermal neutron, and in the conditions at which the reactor is at full power operation with all control rods outside the core.

2.1.1 The control rod drop hydrodynamic analysis

In a study made by Rabiee and Atf [2], the flow field of fluid around control rod inside the guide tube were analyzed by using averaged Navier-Stokes equations and also dynamic mesh technique. In this study, the simultaneous translational and angular velocity of flow fields including moving boundaries and objects are investigated. The FLUENT software was used where the motion of body determined by using a six degree of freedom software module.

The control rods were inserted into guide tubes within fuel elements as in Figure 1. Figure 2 illustrates the control rod motion inside the guide tube. The control rod is consists of two parts with different densities 1.7 and 4.9 g/cm3. The outer diameter and length of control rod are 7.57 mm and 3.55 m respectively. A drain hole near the bottom part of the guide tube simulates the leakage of the tube. Due to leakage of the guide tube, the 15.5 MPa pressure of water moves slowly from bottom to top of the core.

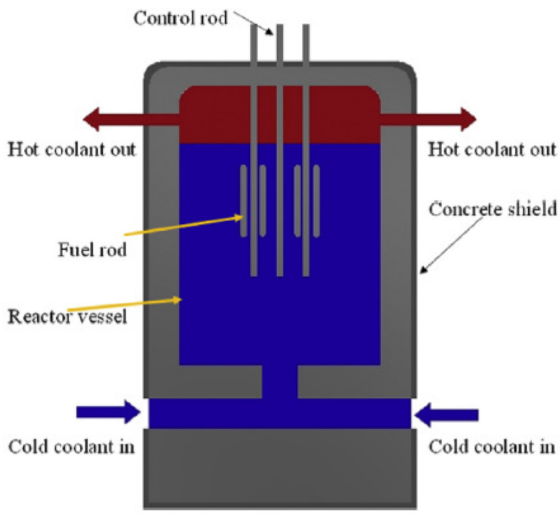

Fig.1. Schematic of control and fuel rods in the reactor core

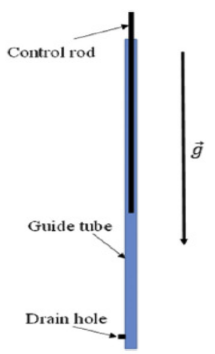

Fig. 2. Schematic of control rod motion inside the guide tube

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The dynamic mesh strategy was used to stimulate the two dimensional axisymmetric flow fields. There are two regions illustrated in computational grid, which are stationary region and the region moves along with control rod (Figure 3).

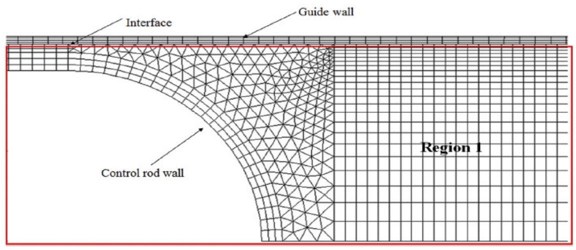


Fig.3. Schematic of the boundary conditions around the control rod including interface

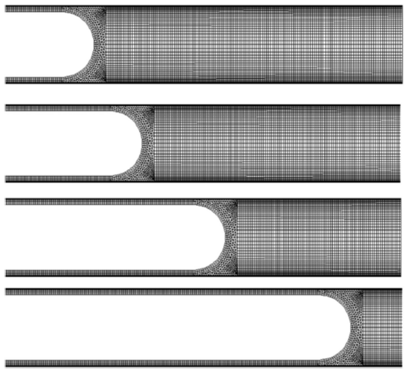


Fig.4. Sequence of the motion of the control rod along the guide tube

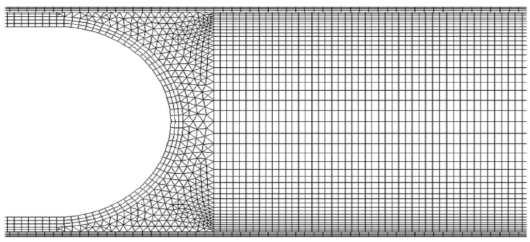


Fig.5. Sequence of the motion of the control rod along the guide tube

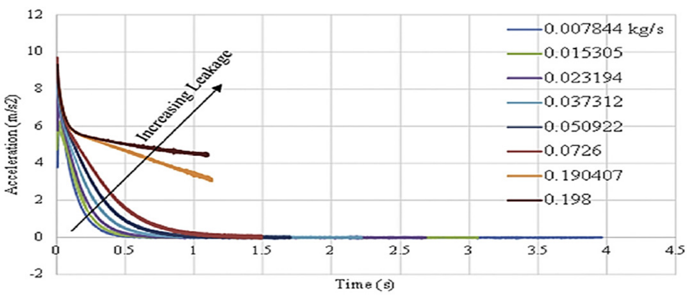


Fig.6. Control rod acceleration at different flow rates versus time

The computational grid around control rod gradually moves away from its initial position along gravity direction as shown in Figure 4. There was also no distortion observed from the computational grids in Figure 5 when two regions, along the interface, move on one another.

2.2 Heterogeneous control rod in Gas-Cooled Fast Reactor (GFR)

The Technology Roadmap of the Generation IV International Forum (GIF) has identified six innovative reactor systems for further development, which are the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Sodium cooled Fast Reactor (SFR), the Supercritical Water Cooled Reactor (SWCR), the Very High Temperature Reactor (VHTR) and the Molten Salt Reactor (MSR) [13].

Since the operation in the fast neutron spectrum gives potential for minor actinide transmutation and waste utilization, the Sustainable Nuclear Energy Technology Platform (SNETP) selected SFR as the major and LFR and GFR as alternative technologies for further development in the European Union [14]. Even though the SFR development can benefit from a more mature technology, GFR must be seen as an alternative technology which could bring extra benefits such as heat production for the industry. The GFR 2400 reactor is the conceptual design of the large scale Gas-cooled Fast Reactor. Its design is based on the initial 600 MWth GFR core proposed by CEA [15] and on the concepts and findings of the EU GoFastR project [16]. Fast systems are characterized by a high migration area of neutrons, thus the reactivity control systems must be based on special principles. Due to the smaller absorption cross section of boron in fast spectrum compared to in thermal spectrum; the neutron reflection outside the core of a fast reactor plays an important role. As a result of the relatively high Pu content in the GFR 2400 reactor core, the effective fraction of delayed neutrons is quite small, while the power density of the core reaches 100 MW/m3 [17]. Therefore the design and operation of the GFR 2400 reactivity control system is a very challenging issue and attention should be paid in order to evaluate the effectiveness of the system of control rods. Due to the early development stage of the pin type GFR 2400 core, no final geometry design of the system of control rods is available other than its homogenous material composition was published [18]. In general, the reactivity worth of control rods (CR) depends on the shadowing effects between individual pins of a single assembly and also mutually between assemblies. Based on Girardin *et al.*, [19] where the GFR 2400 plate type core was investigated it can be assumed that the worth of the final heterogonous CRs will be less than 90% of the worth of the homogenous design. The reactivity worth of CRs can be compensated by adjusting the Boron-10 enrichment of the absorber material which can be done by softening the neutron spectrum, which could also increase the absorption rate of Boron-10 isotopes.

2.2.1 The heterogeneous of control rod design based on a single pin model

In table 1 shows the basic design parameters of the GFR 2400 reactor core. The reactivity of the GFR 2400 core is controlled through two systems of control assemblies. The control (CSD) and diverse safety devices (DSD). The CSD system will be used mainly for reactivity control in normal operation (to achieve criticality by adjusting their position) and the DSD system only for emergency reactor shutdown. The total height of CSD and DSD assemblies is 500 cm, consisting of a 150 cm absorber and a 350 cm follower part. The across flat of both CSD and DSD assemblies is 17.83 cm. The absorber part consists of B4C with 90% weight content of the main absorbing 10B isotope. The rod follower is made of SiC structural material. Since the final pin type GFR 2400 control assembly design was not available, only the homogenous configuration of material parts was used as a starting point of this analysis (denoted as “reference homogenous design”).

Table 1
Basic design parameters of the GFR 2400 reactor core [18-20]

Parameter	Value	Parameter	Value
Reactor thermal power	2400 MW_{th}	Fuel pin diameter	3.335 mm
Gross electric efficiency	45%	Fuel + cladding diameter	4.550 mm
Primary coolant medium	100% He	Fuel pin pitch	11.570 mm
Secondary coolant medium	20% He +80% N_2	Fuel assembly can thickness	2.000 mm
Primary coolant pressure	7 MPa	Fuel assembly pitch	8.915 cm
Prim. coolant mass flow rate	1213 kg/s	Active fuel length	1.650 m
Core inlet/outlet temperature	400/780 ˚C	Number of pins per fuel As.	217 pcs
Core pressure drop	0.143 MPa	Number of fuel assemblies	516 pcs
Secondary coolant pressure	6.5 MPa	Number of CSD/DSD As.	18/13 pcs

The heterogeneous control rod design was proposed based on a single pin model. Pin type arrangement of absorber material is the most common control assembly design for fast reactors. In corporate with the design, the significant consideration is the assessment of the optimal pin radius. For example, if the radius is too large, the absorber material is not utilized in the most effective way, due to self-shielding. For neutronic analysis, neutron penetration distance performance on a simple absorber pin is the best way to find the optimal pin radius. A simple pin design of the GFR 2400 control assembly was created on the basis of the homogenous material compositions. In order to maintain the total volume of material in the assembly, the radius of the B4C absorber (rabs) was calculated to be 5.16 cm and the total radius of the pin (rpin), including the cladding (AIM1) and the helium gap, was 6.28 cm. The exterior covering of the absorber assembly, with an across flat (Hout) 17.53 cm, was made of SiC structure material. Helium was filled in the remaining part of the assembly. The across flat of the helium filled part of the assembly (Hin) was 15.50 cm. A steady state MCNP5 calculation was performed, to calculate the energy integral neutron fluxes as a function of pin radius. The plot of the results is shown in Figure 7.

From the Figure 7, it can be seen that in the exterior area of the B4C absorber, the flux is deteriorated rapidly, but in the area between 0 and 2 cm showed no change of the neutron flux. This phenomenon causes by the geometry self-shielding of the neutron flux. Due to absorption of low energy neutrons at the outer region of the absorber pin, the radius of the studied pin showed a significant value compared to the penetration distance of the neutrons in the absorber. Therefore, the 10B atoms in the central part of the pin have no reasonable chance for neutron capture. Based on Girardin *et al*. [19], it was assumed that in the GFR 2400 control assembly the absorber pins are organized in regular triangular lattice. For the given number of rings Nrings the number of pins Npin can be calculated by Eq. (1). For constant volume of the absorber material Vabs and constant absorber pin height hpin, the pitch of the absorber pins pabs in triangular lattice can be calculated by Eq. (2) and the radius of the absorber pins rabs by Eq. (3). In Eq. (2) Hint is the internal across flat of the assembly cover. The design of the reactivity control system should be as simple as possible, thus the number of absorber pins in the assembly should be minimized.

$$N_{pin} = 1 + \sum_{i=1}^{N_{rings}} 6i \tag{1}$$

$$P_{abs} = \frac{H_{int}}{\sqrt{3}(2N_{rings}-1)} \tag{2}$$

$$r_{abs} = \sqrt{\frac{V_{abs}}{N_{pin} h_{pin} \pi}} \tag{3}$$

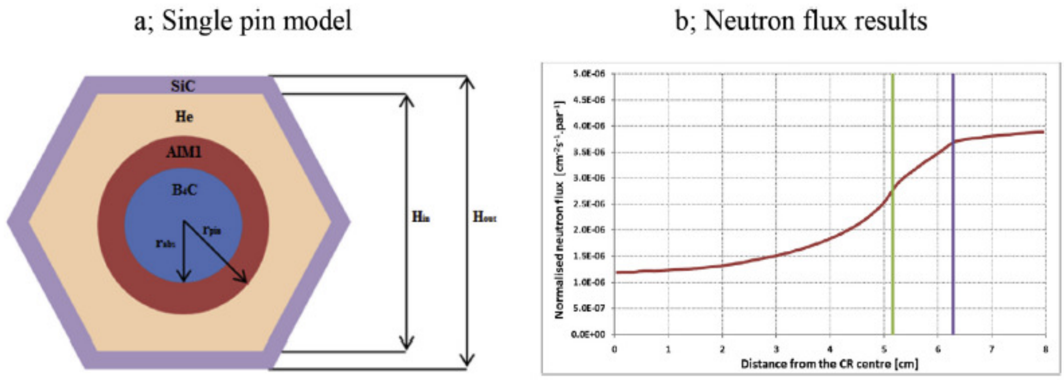


Fig.7. Cross section of the single pin GFR 2400 control rod and flux results. [21]

To judge the CR design in terms of economy, the C2 coefficient was introduced (Eq. 5). This coefficient represents the normalization of the C1 coefficient per number of absorber pins (Npin). The neutron fluxes at given equivalent pin positions, ϕ (req) were compared by means of the C1 coefficient defined by Eq. (4). In this sense less absorber pins mean simpler and economically reasonable design.

$$C_1 = \frac{\varphi(r_{eq})}{r_{abs}\varphi_0} \tag{4}$$

$$C_2 = \frac{1E3C_1}{N_{pin}} \tag{5}$$

2.2.2 Improving the performance of control rods by spectral shifting

There are several options how the worth of a control rod could be increased. The simplest option would be to increase the mass content of B4C in the CR, which would however make their construction more expensive. In addition, by putting more material to a given volume of CR the size of the absorber pins would increase and a significant self-shielding effect could occur. It is also possible to change the 10B enrichment or the 10B/11B ratio in the absorber pins, but in the current design the 10B enrichment is still 90% thus further enrichment would not bring the expected benefit. Due to neutron moderation on 11B atoms the neutron spectrum could be slightly shifted towards lower energies, were neutron capture on 10B is more likely. Unfortunately, by decreasing the 10B/11B ratio the 10B content would decrease which would compensate the increased in absorption cross sections. Another promising option is to accommodate extra moderator material in the CR assembly. By making the neutron spectrum softer, the absorption rate on 10B would increase. This would also increase the reactivity worth of the CR assembly. The neutron spectrum could be shifted by either elastic or inelastic reactions, however in this paper the emphasis was put on selecting only material which are characterized by high elastic scattering cross section. Materials used as extra moderators in the CR design must not be subjected to phase changes, must keep solid state and must withstand the radiation damage caused by fast neutrons. After comprehensive study of potential candidate materials (Cerba et al., 2013), ZrH2.0, ZrH1.5, natural LiH, enriched LiH, BeO, MgO, Li2O and SiC were selected for investigation. The basic parameters of selected moderator materials are shown in Table 2.

Table 2
Basic parameters of the selected moderators [22]

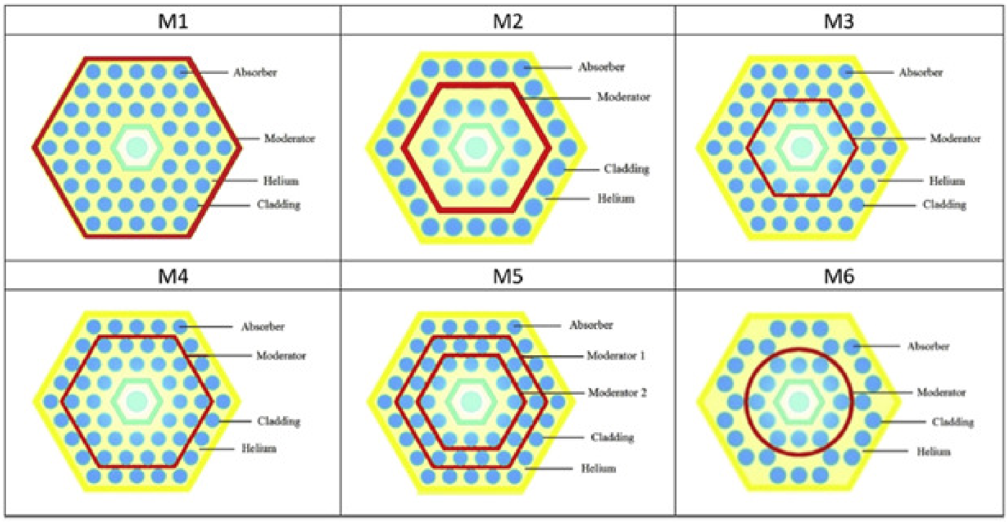
Material	Abundance	Density [g.cm^{-3}]	
$ZrH_{1.5}$	Natural	5.56	1073.16
$ZrH_{2.0}$	Natural	5.56	1073.16
LiH	Natural	0.78	961.86
LiH	Enriched ^{7}Li	0.78	961.86
BeO	Natural	3.01	2780.16
MgO	Natural	3.58	3125.16
Li_2O	Enriched ^{7}Li	2.01	1711.16
SiC	Natural	2.60	3003.16

From the table, BeO, MgO, Li2O and SiC show high resitance to high temperature. For each moderator 6 geometry configurations (M1-M6) were investigated. The basic parameters related to the geometry models are shown in Table 3 and the horizontal cross sections of the created CR assemblies are in Table 4. Evaluation of the moderator effectiveness was calculated based on equation shown [21].

Table 3
Basic parameters of control rods with moderators [21]

Modification	**M1**	**M2**	**M3**	**M4**	**M5**	**M6**
Shape	Hex	Hex	Hex	Hex	Hex	Circle
N_{pins}[-]	54	36	54	54	54	36
r_{abs} [mm]	6.08	7.02	6.08	6.08	6.08	7.02
V_{abs} [l]	13.79	9.19	10.34	10.34	10.34	9.19
N_{mod} [-]	1	1	1	1	2	1
H_{mod}[mm]	6	5	3	3	3 + 3	3
V_{mod}[l]	2.88	3.29	1.56	2.18	3.74	1.80
$\frac{100 \cdot V_{mod}}{V_{abs}}[\%]$	20.90	35.76	15.09	21.06	36.15	19.59

Table 4
Horizontal cross section of the control rod modification [21]



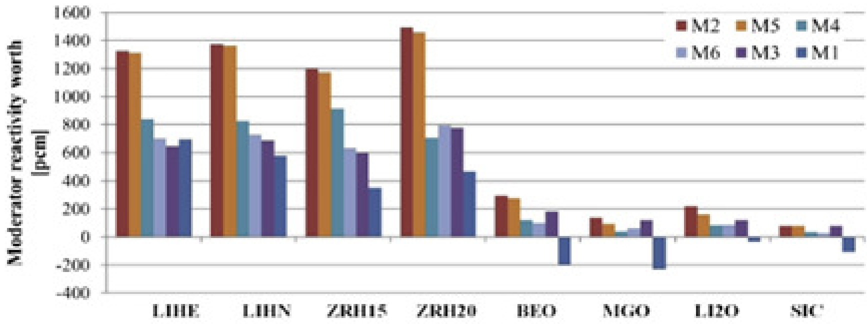


Fig. 8. Results of the moderator reactivity worth [21]

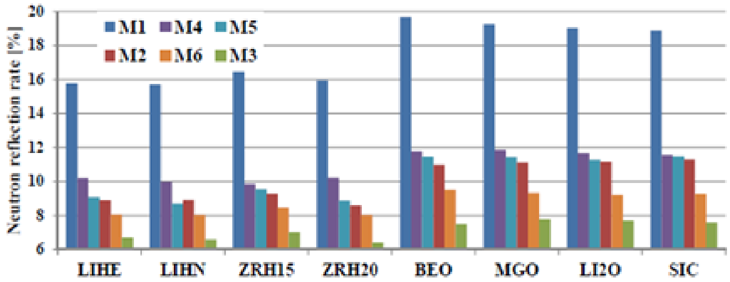


Fig. 9. Results of neutron reflector rate in the IFIR region per moderator material [21]

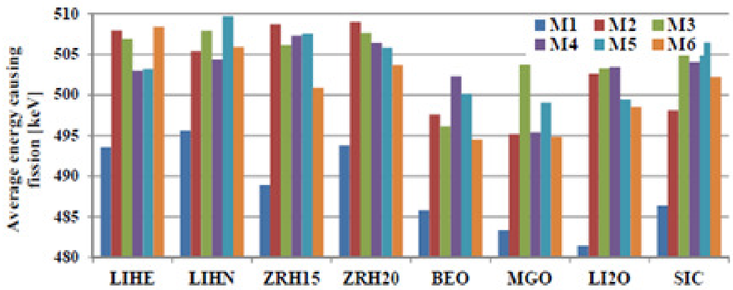


Fig. 10. Results of the average energy causing fission in the IF1R region per moderator material [21]

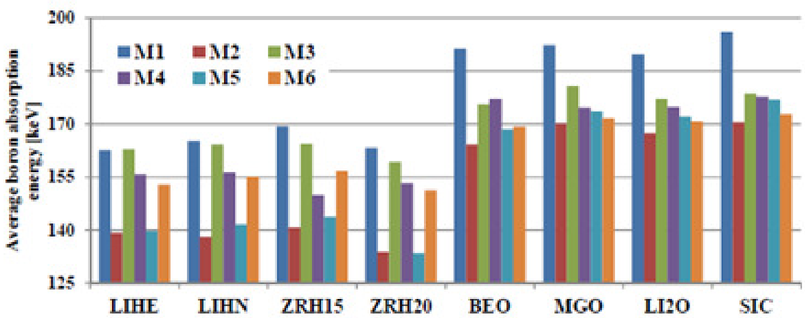


Fig.11. Results of the average boron absorption energy per moderator material [21]

For this heterogeneous design the reactivity worth of 91.3% of the homogeneous design was achieved, as shown in Figure 8 until Figure 11. The possibilities to increase the CR worth by increasing the efficiency of the neutron absorber were studied on 6 modifications of the heterogeneous CRs using 8 moderator materials. Emphasis was put on the selection of appropriate moderator materials and geometries to optimize the CR worth, the neutron reflection rate, the boron absorption energy and the energy of neutrons causing fission in the surrounding region. It was found that the moderator to fuel distance plays important role in optimizing the reflection rate and the energy of neutrons scattered from the moderator to the fuel region. Due to this fact the moderator segment should not be placed at the periphery of the control rod. To enhance neutron absorption on boron, the neutrons should be slowed down. It was also found that the extent of neutron moderation depends more on the volume and type of moderator material than on its geometry configuration. In all of the investigated cases the best properties were found for the hydride moderators, $ZrH2.0$, $ZrH1.5$ and LiH. Unfortunately these materials have undesirable properties at high temperatures which could lead to hydrogen and helium production, therefore hydrides are not recommended. Although BeO and $Li2O$ showed worse properties but their impact on the control rod could be enhance by increasing their mass proportion.

2.3 Safety control rod for shutdown purpose in Miniature Neutron Source Reactor (MNSR) GHARR-1

In some countries, their Low Power Reactors (LPR) with simple instrumentation and control may have a problem of licensing. This may be the reason why single control rod is used for the Slowpoke and MNSR, thereby challenging some of the safety criteria [23]. Therefore, the idea to construct an additional safety rod for the MNSR will contribute as the solution of this problem. Besides that, reactor plant has developed from the early work where the aim was to demonstrate that a nuclear reactor was feasible to the present time where safety is a foremost consideration [24]. The past records have shown that the philosophies used as the current safety practiced in the design of nuclear power plants and their operations are adequate enough. However, a backup plan is always needed for additional safety measures [25]. Experiments done at the research reactor have significantly contributed to the safety analysis of current and future power reactors.

2.3.1 Modelling and simulation of the input deck

The technical and the design specifications of reactor cores is based on the interaction of a few variables including fuel type, the reflector and moderator, core geometry and composition, radiation shielding, thermal hydraulics, criticality and reactor safety.

The additional safety rod was considered as a cylindrical rod that consists of an absorber section with stainless steel clad of thickness 0.055 cm. Two different absorber materials were model for two different cases, which is cadmium and boron carbide. For the first case, the absorber rod was model with radius cadmium is 0.1950 cm, same as the radius of the absorber material in the present central control rod. The rod radius was then increased sequentially by 0.1 cm, with the clad thickness maintain until it was 0.9500 cm, with the radius of the Cd reaching 0.8950 cm and the rod filling about 86.63% of the channels volume. After that, the irradiation channel was getting higher to a height of 22.6375 cm and radius of 1.7 cm in order to accommodate a bigger cadmium safety rod (LCSR). The SR was appropriately increased in width to a radius of 1.55 cm and height of 36.53 cm, covering about 90% of the channels volume. The process was repeated with boron carbide as the absorber material but without the further increment in the safety rod and the irradiation channel housing it. On simulation, continuous-energy cross-section data were used. All calculations were made using the full spectrum of energy available at the MCNP5 code library at 20 ◦C. The effective multiplication factor was computed in MCNP5 based on the calculation of three different estimators; a collision-based keff, an absorption-based keff, and a track-length-based k_{eff} [26]. The input file for the MCNP5 included 450 cycles made of 50 inactive cycles and 400 active cycles with 500,000 histories per cycle. It took on the average 7628.43 min on an Intel® CoreTM 2 Quad CPU Q6600@ 2.40 GHz, 4.00GB RAM, 32 bit operating system to run each input.

2.3.2 Calculation of safety parameters for the reactor model

To maintain a safe shutdown position, the safety parameters was considered in this work where the effective multiplication factor (k_{eff}), flux level (φ), safety rod worth, shut down margin (SDM) and safety reactivity factor (SRF). MCNP does the necessary criticality computations and outputs the final k_{eff} value with its deviation in the tally. MCNP also calculates the fluxes and outputs them in the tallies in not normalized format. The fluxes were then normalized with the following relation:

$$\text{Flux} = \frac{3.450908\text{E} + 10 \times P(\text{W}) \times \text{flux tally}}{\text{loss to fission x Vol.of channel segment}} \tag{6}$$

The reactivity worth of a safety rod, can then be calculated from equation below:

$$\text{Reactivity} = \frac{\text{k0} - \text{k}}{\text{k0 x k}} \tag{7}$$

where, k_0 is the k_{eff} before rod insertion and k is the k_{eff} after rod insertion. The SDM is the differences between the rod reactivity worth and the core excess reactivity.

Shutdown margin = Rod worth − core excess reactivity (8)

The SRF is defined as the ratio of the calculated reactivity worth of the rod to the core excess reactivity.

$$\text{SRF} = \frac{\text{calculated reactivity worth of the rod}}{\text{core excess reactivity}} \tag{9}$$

2.3.3 Effects of the safety rods on k_{eff}

According to Anim-Sampong *et al*. [27] and as confirmed by GHARR-1 SAR, a criticality safety criteria established in the operating limits and conditions (OLC) for the GHARR-1 facility permits a shutdown margin as low as 1.5 mk of reactivity. Plus, as indicated by Balogun [28], the IAEA requires that the SRF should be bigger than 1.5. As shown in Table 3, the core gets subcritical (0.99665 ± 0.00006) when the CCR is inserted. Its reactivity worth of SDM of 3.37 mk and 7.38 mk are both comparable to the SAR's value of 6.8 mk and 3.0 mk, respectively with the rod worth being 8.5% higher and the SDM 12.3% higher. The SRF of 1.84 is also within the limit set by the IAEA. When the CCR is stuck in the withdrawn position or malfunctioning, the SRs will then be inserted to shut down the reactor. On insertion of the CSR of reactivity worth 2.94 mk, the keff of 1.00106 ± 0.00006 means the reactor is still critical. Therefore, this model will not attain an SDM and its SRF of 0.74 is also below the IAEA's limit. The LCSR of reactivity worth 4.54 mk (38.5% less the reference) on insertion, reduces the keff to 0.99775 ± 0.00007 (subcritical) and an SDM of 2.25 mk, which in the acceptable limit. The SRF of 1.98 is also within the IAEA's acceptable limit. The last of the SRs, B4CSR attains a keff of 0.99803 ± 0.00006 (subcritical) when inserted. Its reactivity worth is 5.90 mk, SDM after insertion is 1.97 mk (41.5% less the reference value) and SRF of 1.50.

Table 5
Table of safety parameters for the central control rod and the safety rod models in inserted position (shut down mode) [29]

Type rod	Effective multiplication factor (fully inserted) k_{eff}	Computed rod worth (mk)	Shut down margin (SDM) (mk)	Safety reactivity factor (SRF)	Criticality
CCR	0.99665 ± 0.00006	7.38	3.37	1.84	Subcritical
CSR	1.00106 ± 0.00006	2.94	-	0.74	Critical
LCSR	0.99775 ± 0.00007	4.54	2.25	1.98	Subcritical
B_4CSR	0.99803 ± 0.00006	5.90	1.97	1.50	Subcritical

2.3.4 Effects of the safety rods in the inner irradiation channels on the average neutron flux distribution

The control or safety rods are tends to absorb neutrons when it is inserted, thereby reducing the neutron population. The CCR with a worth of 7.38 on insertion, reduces the average neutron flux to (5.921 ± 0.0018) E+11 n/cm2 s, i.e. 47.3% reduction. The average neutron flux of (8.248 ± 0.0017)E+11 n/cm2 s was recorded in inner irradiation sites 1, 2, 4 and 5 with site 3 when the CSR was inserted, which houses the safety rod, recording (2.354 ± 0.0018)E+11 n/cm2 s. The CSR therefore decreases the average neutron flux by 36.8% on insertion. Similar trends were recorded for the LCSR and B4CSR with the LCSR recording low fluxes of (2.991 ± 0.0018) E+11 n/cm2 s and B4CSR, (6.510 ± 0.0018) E+11 n/cm2 s in inner irradiation sites 1, 2, 4 and 5. In both cases, site 3 recorded very low fluxes of (1.647 ± 0.0018) E+11 n/cm2 s and (1.712 ± 0.0018) E+11 n/cm2 s for LCSR and B4CSR, respectively [29].

3. Conclusion

For the safe operation of a reactor, the operating organization should have adequate information on the control rods and on the safety handling procedure of the control rods. The results shows boron carbide that have been used in the safety rods as absorber material had a better shut down parameters. The best properties for heterogeneous design were found for the hydride moderators, ZrH2.0, ZrH1.5 and LiH, unfortunately these materials have undesirable properties at high temperatures but, the BeO and Li2O moderators showed worse properties but their impact on the control rod could be enhanced by increasing their mass proportion in the assembly. Event that cause by the uncontrolled reactivity often related to the rod drop accident in the reactor safety analysis. The study of the rod drop are mainly to hydro-dynamically evaluate the control rod drop reactor during the accident. This study also shows that by decreasing leakage flow rate and in certain leakages, the total force exerted on the control rod would be reduced.

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