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NEUTRONIC SAFETY ANALYSIS OF PROPOSED REACTOR TECHNOLOGIES FOR GHANA'S NUCLEAR POWER PLANT USING THE MCNP CODE

by

Rex G. ABREFAH [1,2*], ***Prince M. ATSU*** [1], ***and Robert B. M. SOGBADJI*** [1,3]

[1] Graduate School of Nuclear and Allied Sciences, University of Ghana, Accra, Ghana,
[2] Ghana Nuclear Regulatory Authority, Atomic-Kwabenya, Ghana
[3] Ghana Atomic Energy Commission, Nuclear Power Institute, Legon, Ghana

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In pursuance of sufficient, stable and clean energy to solve the ever-looming power crisis in Ghana, the Nuclear Power Institute of the Ghana Atomic Energy Commission has on the agenda to advise the government on the nuclear power to include in the country's energy mix. After consideration of several proposed nuclear reactor technologies, the Nuclear Power Institute considered a high pressure reactor or vodo-vodyanoi energetichesky reactor as the nuclear power technologies for Ghana's first nuclear power plant. As part of technology assessments, neutronic safety parameters of both reactors are investigated. The MCNP neutronic code was employed as a computational tool to analyze the reactivity temperature coefficients, moderator void coefficient, criticality and neutron behavior at various operating conditions. The high pressure reactor which is still under construction and theoretical safety analysis, showed good inherent safety features which are comparable to the already existing European pressurized reactor technology.

Key words: reactivity temperature coefficient, moderator void coefficient, criticality, neutron behavior

INTRODUCTION

In the midst of two crucial global challenges, climate change and the growing demand for energy, many countries around the world are working towards meeting the energy demands with next to zero carbon emissions. Nuclear energy offers a very significant pathway out of the challenges. Nuclear energy technologies are capable of producing larger amounts of energy which contributes immensely to meeting the high energy demands.

Ghana is one country looking to add nuclear energy to its energy mix. The nuclear power program under the Nuclear Power Institute of the Ghana Atomic Energy Commission has made significant inroads in that regard, proposing two-reactor technologies. The high pressure reactor (HPR) and the vodo-vodyanoi energetichesky reactor (VVER) are the two-reactor technologies proposed for Ghana's first nuclear power plant.

As part of a technology assessment, the neutronic safety parameters of both reactor technologies are investigated and compared. Safety analysis is also crucial in aiding countries to make decisions on the type of reactor systems to build and also to incorporate the lessons learned from nuclear accidents into safety analysis for added safety assurance [1]. Knowledge of changes in reactivity caused by changes in void content and temperature are necessary. Reactivity coefficients are important for reactivity and power excursion transient analysis. The sign, rate of the change, response time, and magnitude of reactivity coefficients are of great importance [2].

* Corresponding author; e-mail: adibola2001@yahoo.com

DESCRIPTION AND COMPARISON OF REACTORS TECHNOLOGIES

Neutronic safety in reactor operation is affected by the configuration and distribution of fuel assemblies and the fuel enrichment in the reactor core. The configuration of the reactor core depends on the geometry, dimensions and material composition of the reactor core.

The HPR-1000 was designed by the China Zhongyuan Engineering Corporation under the supervision of the China National Nuclear Corporation [3]. The HPR1000 reactor core generates 3050 MW of thermal power with an average linear power density of 173.8 Wcm^{-1}[4]. The reactor core is loaded with 177 China fuel series (CF3) fuel assemblies, ensuring sufficient thermal margin while increasing output power. The CF3 fuel assembly (FA) is composed of 264 fuel rods arranged within a 17 × 17 supporting structure. The fuel rods contain UO_2 pellets or Gd_2O_3-UO_2 pellet [4]. Zircalloy is used as a cladding material for the fuel pins. The CF3 has excellent performance and is applicable for a long refueling cycle. Three independent means exist for core reactivity and power distribution control: burnable absorber of gadolinium (Gd_2O_3) poisons, rod cluster control assemblies (RCCA), and soluble boron absorber. The RCCA is comprised of 24 control rods fastened to a spider connector. The absorber material used in the control rod is Ag-In-Cd alloy or stainless steel. The HPR1000 is designed with a thermal margin greater than 15 % to improve safety and operational performance [4]. The HPR is still under construction.

The VVER reactor was developed by ROSATOM subsidiary OKB Gidropress, while the nuclear power stations employing the VVER have been developed by the power plant design organizations within ROSATOM: Moscow Atomenergoproekt, Saint-Petersburg Atomenergoproekt (a branch of VNIPIET), and Nizhniy Novgorod Atomenergoproekt [5]. The VVER-1000 reactor core is comprised of an array of 163 hexagonal fuel assemblies with an active core height of 3.53 m. The fuel assemblies are identical in geometrical design but are different in fuel enrichment, based on the position of each assembly within the reactor core. The lattice pitch is 23.6 cm. The fuel rods are arranged in a hexagonal structure inside FA. The advanced nuclear fuel for reactors of Russian design (TVSA) FA is considered as a base version of FA design and as an alternative version to the TVS-2. Both versions of FA are interchangeable and are of reference character. The core design is developed for the generalized version of FA design (both base and the alternative) providing its operability in using several FA types. Bundles of fuel rods, fuel rods, and gadolinium (Gd) fuel rods, consist of a skeleton that houses 312 fuel rods (Gd fuel rods). The VVER-1000 reactor core generates 3000 MW of thermal power. The rod control cluster assembly (RCCA) consists of 18 absorbing elements (AE), with Boron carbide and Dysprosium titanate (B_4C and Dy_2O_3 TiO_2) used as absorbing material. Dysprosium titanate in the AE lower part enables to extend RCCA service life under maintenance of sufficient worth of emergency protection.

THEORY

Void coefficient of reactivity

In water-moderated reactors, changes in moderator density significantly affect the reactivity. Changes in moderator density can be due to thermal expansion, void formation or loss of coolant. A change in the moderator void content leads to a change in multiplication factor, k, and alters the reactivity of the system. The void coefficient of reactivity is therefore defined as the rate of change in the reactivity of a water-moderated reactor resulting from any modification of the moderator/coolant as the power level and temperature changes. The principal effect is the loss of moderation that accompanies a decrease in moderator density and causes a corresponding increase in resonance [6]. For pressurized water reactors, about 80 % of neutron moderation occurs in the light water moderator.

For a given value of k_∞, the reactivity, ρ, in the core is determined by the expression

$$\rho \quad \frac{k_0 \quad k_\infty}{k_0} \tag{1}$$

$$\Delta\rho \quad \rho_0 \quad \rho_1 \tag{2}$$

where k_∞ is the multiplication factor at the present reactor operating condition and k_0 is the multiplication factor at the normal reactor operation condition. From the definition of the void coefficient of reactivity given in eq. (1), the void coefficient of reactivity, $\gamma\xi$, is mathematically given as

$$\gamma\xi \quad \frac{\Delta\rho}{\Delta\xi} \tag{3}$$

The ξ represents the reactor parameter affecting the reactivity and $\Delta\rho$ representing the corresponding change in reactivity. If the ξ represents void, then the change in the void is $\Delta\xi$ and the void coefficient of reactivity is defined by $\gamma\xi$.

TEMPERATURE COEFFICIENT OF REACTIVITY

The influence of temperature on the neutron transport is caused by the thermal movement of nuclei influencing the scattering of thermal neutrons and the Doppler broadening of resonances which is due to variation in neutron cross-section with temperature and by the thermal expansion of different materials within the core. Two main temperature coefficients are defined in respect to which temperature change is considered: fuel temperature reactivity coefficient, moderator temperature reactivity coefficient. Reactivity changes associated with a degree change in the moderator temperature is referred to as the moderator temperature coefficient of reactivity [7]. The value of the temperature coefficient is determined by simply dividing the change in reactivity $\delta\rho$ due to change in temperature by the corresponding change in temperature δT [8]

$$\alpha \quad \frac{\delta\rho}{\delta T} \tag{4}$$

The temperature coefficient of reactivity, α, has different effects on reactivity in the core:

– a nuclear temperature coefficient arising from a change in cross-section with changing neutron temperature,
– a density temperature coefficient arising from a change in temperature, and

– volume temperature coefficient arising from a change in geometric buckling when the temperature changes.

The total temperature coefficient of reactivity is given by the sum of the moderator temperature coefficient and fuel temperature coefficient

$$\alpha \quad \frac{\delta\rho}{\delta T}_{\mathrm{f}} \quad \frac{\delta\rho}{\delta T}_{\mathrm{m}} \tag{5}$$

$$= \alpha_{\mathrm{f}} + \alpha_{\mathrm{m}} \tag{6}$$

The reactivity change is given by eq. (1) and (2).

MODELLING AND REACTIVITY COEFFICIENT CALCULATIONS

To calculate the reactivity coefficients of the core of the reactors, a detailed 3-D computational model of the HPR and VVER cores were originally developed using the Monte Carlo neutronics behavior simulating code MCNP5. The code utilized the ENDF-VI as a cross-section library for the materials in these computations. The fuel assemblies (figs. 1 and 2) were modelled to include all fuel pins in their respective lattice and the moderator/coolant for each reactor core. Materials and their various compositions were specified in the code. The significant difference in the HPR and the VVER MCNP model is the geometry, lattice, number of pins and moderator fuel ratio.

In calculating the effects of void on the reactivity in the reactor core, the developed MCNP input model was modified to have different moderator densities to depict increasing void content whiles other conditions in the core were kept constant. The core temperature was also varied whiles other conditions were kept constant in other to determine the effect of the changing temperature on the reactivity in the core of the reactors.

These calculations were carried out with a total number of 550 cycles of iteration on a source size of 500 000 particles per cycle. The first 50 cycles were skipped to decrease statistical errors in the estimates. The k_{eff} was obtained for each run from the respective output to calculate the corresponding change in reactivity.

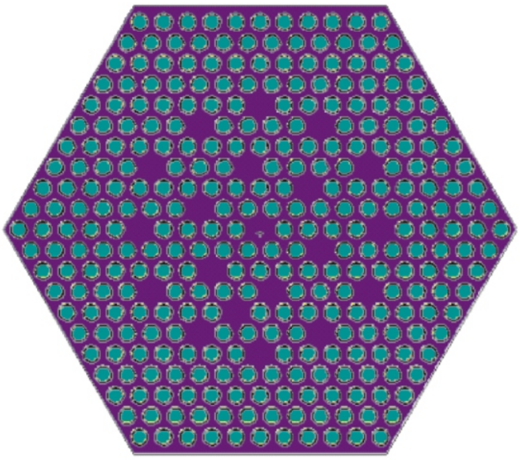

Figure 2. The MCNP plot of VVER fuel assembly

The MCNP neutron energy spectrum was performed for 20484 energy grids combined for all three categories of the energy distribution: thermal, slowing down, and fast. The following energy bins were used in the MCNP tally for the various energy groups: $1.89\ 10^{-08}$ MeV energy bin for 0 to $6.25\ 10^{-07}$ MeV thermal energy range, $1.89\ 10^{-03}$ MeV energy bin for 0.821 to 6.94 MeV slowing-down energy range, and $1.89\ 10^{-03}$ MeV energy bin for 6.96 to 20 MeV fast energy range.

RESULTS AND DISCUSSIONS

Increasing void in the reactor core causes the moderator to expand, increasing the space within the water molecules and effectively reducing the probability of interaction between the fission neutrons and the atoms of the moderator. When this happens, thermalization of fast fission neutrons reduces and the rate of the fission reaction also reduces thereby reducing the reactivity in the core. The negative void coefficient of reactivity is shown by both reactors in fig. 3 [9]. This is a safety design feature of all pressurized water reactors. Hower, the neutron spectra of different percentage of loss of coolant for the HPR and EPR assamblies are shown in figs. 4 and 5.

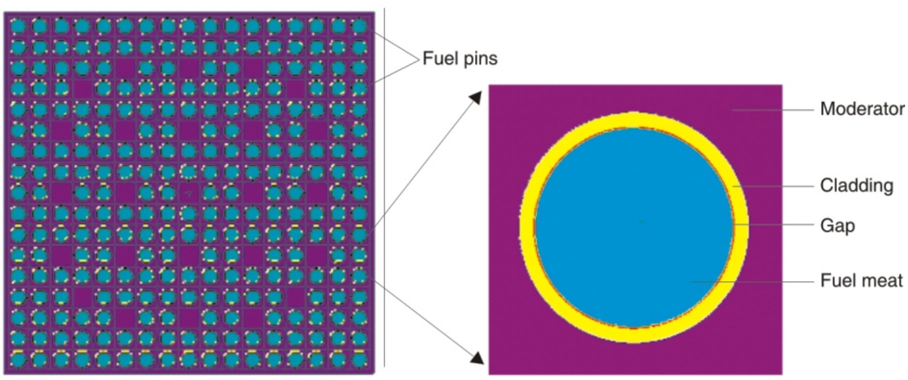

Figure 1. The MCNP plot of HPR fuel assembly

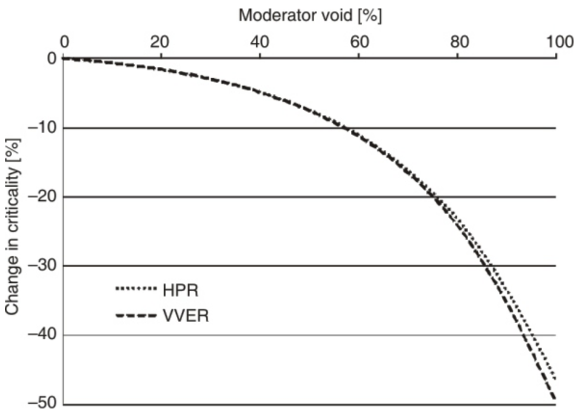

Figure 3. Moderator void coefficient for the HPR and VVER, comparable to Sogbadji [9]

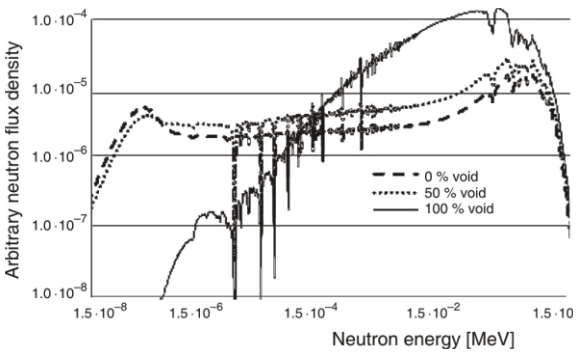

Figure 4. Neutron spectra of different percentage of loss of coolant for the HPR assembly, comparable to Sogbadji [9]

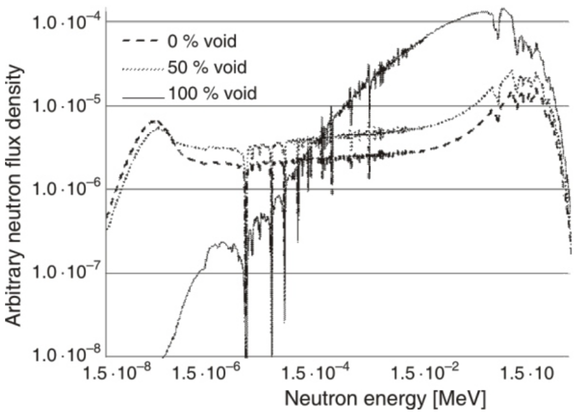

Figure 5. Neutron spectra of different percentage of loss of coolant for the EPR assembly, comparable to Sogbadji [9]

The moderator temperature coefficient of all the reactor assemblies under study is desirable since the reactivity decreases with increasing moderator temperature as shown in fig. 6. An increase in the moderator temperature makes the core under moderated due to the increase in energy of the lighter nuclides that can cause moderation. Nuclei cross-sections are energy-specific hence a change in the energy of the nu-

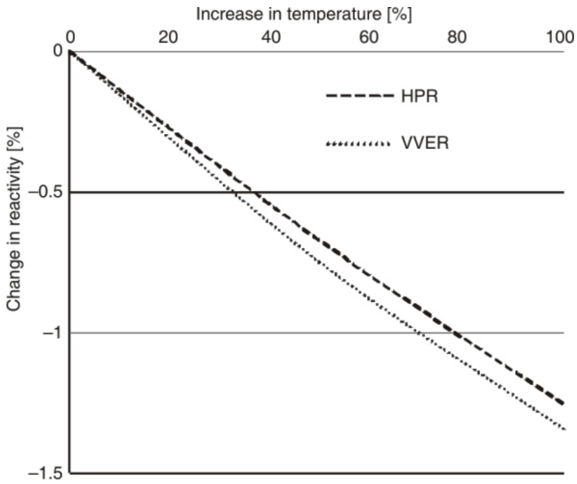

Figure 6. Moderator temperature coefficient of the two reactor assemblies, comparable to Sogbadji [9]

clei changes the probability of interaction. In this case, the increasing temperature does not necessarily reduce the probability of nuclei interaction since the decrease in the resonance peak height is compensated for by the broadened width. An under moderated reactor gives a negative moderator temperature coefficient whiles an over moderated reactor will give a positive moderator temperature coefficient.

CONCLUSION

The criticality and the reactivity changes of the HPR and VVER at various operation conditions were analyzed and compared. The effects of increasing void fraction and increasing core temperature on the reactivity of the reactors and the associated coefficients of reactivity were also calculated. Modelling and simulation of the fuel assemblies of the reactors were carried out using the MCNP5 neutronics code. The MCNP code was used to determine the k_{eff} of the cores at the different operating conditions. Both the HPR and VVER showed good inherent safety of –0.0126 and –0.0134 for temperature, respectively. The moderator void coefficient was also negative for both reactors at all times which is a desired design safety feature for pressurized water reactors.

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AUTHORS' CONTRIBUTIONS

The modeling and calculations using MCNP5 were performed by R. G. Abrefah. The manuscript and the figures were prepared by P. M. Atsu and were revised by R. B. Sogbadji. All the authors contributed to the development and the use of MCNP code, including analyses and discussion of the results.

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Рекс Г. АБРЕФАХ, Принц М. АЦУ, Роберт Б. СОГБАЂИ

АНАЛИЗА НЕУТРОНСКЕ СИГУРНОСТИ ПРЕДЛОЖЕНИХ РЕАКТОРСКИХ ТЕХНОЛОГИЈА ЗА НУКЛЕАРНУ ЕЛЕКТРАНУ У ГАНИ ПРИМЕНОМ MCNP ПРОГРАМСКОГ ПАКЕТА

У потрази за довољном, стабилном и чистом енергијом, којом би се решила постојећа енергетска криза у Гани, Институт за нуклеарну енергију Комисије за атомску енергију Гане има дужност да саветује Владу у укључивању нуклеарне енергије у постојеће енергенте. Након разматрања неколико предложених технологија нуклеарних реактора, Институт за нуклеарну енергију установио је да су реактор са водом под притиском, или VVER реактор, избори нуклеарних технологија снаге за прву нуклеарну електрану у Гани. Као део процена технологије, испитани су параметри неутронске сигурности оба типа реактора. MCNP програмски пакет примењен је као алат за прорачун и анализу температурног коефицијента реактивности, void коефицијента модератора, критичности и понашања неутрона при различитим режимима рада. Реактор са водом под притиском, који је још увек у изградњи, и теоријска анализа сигурности указују на добра инхерентна својства сигурности која су упоредива са већ постојећом европском технологијом реактора под притиском.

Кључне речи: температурни коефицијент реактивности, void коефицијент модератора, критичност, понашање неутрона