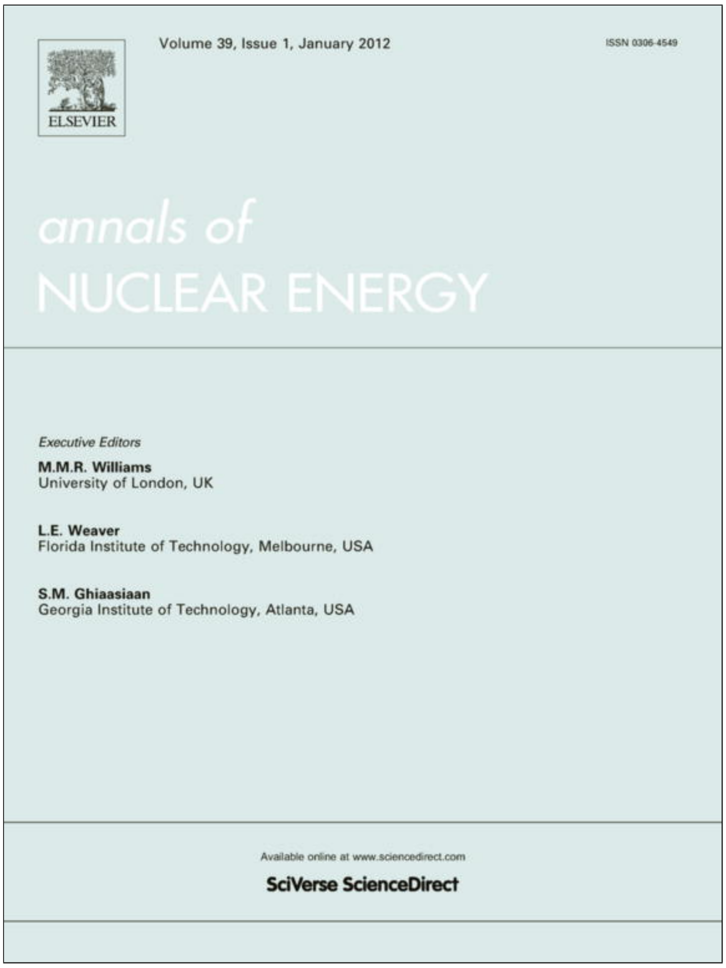
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Monte Carlo simulation of additional safety control rod for commercial MNSR to enhance safety

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ABSTRACT

A Monte Carlo simulation of additional safety rods for NIRR-1 HEU and LEU cores was carried out using the MCNP5 version 1.6 code. Two additional safety control rods having the same material composition as the main central control rod except for the surface area were studied. The following reactor core physics parameters were determined; neutron flux distribution within the core with safety rods withdrawn, control rod (CR) worth for each rod, core excess reactivity, shutdown margin and some kinetic parameters. Results obtained indicate that it would be feasible to include two additional safety control rods to improve safety level of the MNSR with little or no modification to the existing core configuration.

1. Introduction

The Miniature Neutron Source Reactor (MNSR) is a low power reactor, which was designed to have one central control rod that performs safety and regulatory functions (Jonah et al., 2007; SAR, 2005). Control rods are strong neutron absorbers that can be inserted or withdrawn from the core. They are basically used to compensate for the excess reactivity necessary for long term core operation and also to adjust the power level of the reactor in order to bring the core to power, follow load demands and shutdown the reactor (Stamatelatos et al., 2007; Varvayanni et al., 2009).

The design has been carried out in accordance with the Chinese standards with some attention focused on relevant International Atomic Energy Agency (IAEA) standards (IAEA, 2009). However, this design is not fully in conformance with the IAEA safety standards. In particular, the lack of redundancies of control rods and neutronics does not provide for protection against single failure. Because the movement of the single control rod is controlled by mechanical clutches, it is possible for the mechanical systems to malfunction and therefore may not allow the control rod to perform its intended safety function. In the case of malfunction of the single control rod such as rod stuck situation, emergency shutdown of the reactor is achieved by pumping Cd rabbits and strings into irradiation channels. Pumping of cadmium rabbits may not also be achieved in case of pressure failure. This situation may lead to excessive power excursion and ultimately to radiation exposure of personnel in the process of inserting Cd strings in the irradiation channels in order to bring the reactor under control. Even though the MNSR is inherently safe due to high negative temperature coefficient of reactivity, which provides self limiting power excursion characteristics, reactor safety experts have recommended for redundancies in design for reliability of systems important to safety (IAEA, 2005).

In order to address the single point failure of the current HEU core and consequently improve the safety of the MNSR without any modification to the fuel region in the core, the inner irradiation channels of Nigeria Research Reactor-1 (NIRR-1) located at the Centre for Energy Research and Training (CERT), Ahmadu Bello University in Zaria, has been used to introduce two additional safety control rods (ASCR) using the computing facilities of the Reduced Enrichment for Research and Test Reactors (RERTR) program, Nuclear Engineering Division, Argonne National Laboratory, USA. The control rods introduced will work simultaneously and it is proposed to be controlled by a single mechanical device such that when scrammed into the reactor, they would fall under gravity. There have been national and international activities to convert research and test reactors from the use of highly enriched uranium (HEU) to low enriched uranium (LEU) fuel. Achieving the conver-

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sion of all MNSR is helpful in this international effort to reduce and eventually eliminate HEU in civilian use (Travelli, 1999). Recently, the CIAE has successfully performed zero power test of a variant of the MNSR, called the In-Hospital Neutron Irradiator (IHNI) designed exclusively for the treatment of cancer based on the boron neutron capture therapy (BNCT) technique (Ke et al., 2009). The INHI has similar design characteristics with commercial MNSR. With regards to core configuration, the core of INHI is fueled with UO_2 fuel enriched to 12.5% in Zr-4 clad. MNSR operating countries including Ghana, China and Nigeria have been studying conversion of the current HEU core to LEU. Conversion studies of the NIRR-1 began in 2006. An input deck was created for the reactor using the MCNP 4C code. The code was benchmarked against experimental results (Jonah et al., 2007). Feasibility studies for the conversion carried out indicated that it is possible to convert the reactor from 90.2% HEU Al_4 to 12.5% LEU UO_2 fuel (Jonah et al., 2009). Because conversion from HEU to LEU is envisaged, the potential LEU core was also modified with the two additional safety control rods.

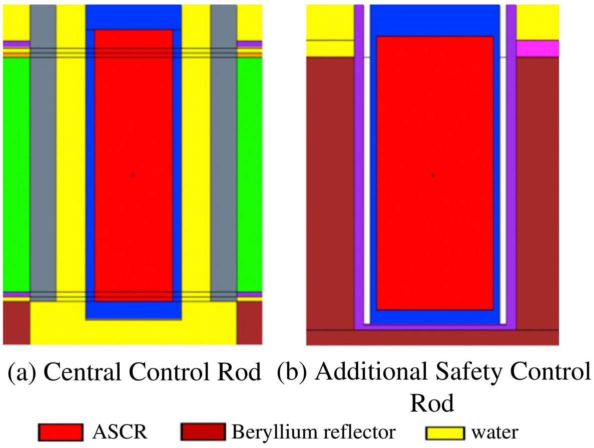

Fig. 2. MCNP geometric representation of central and ASCRs.

2. Description of NIRR-1 reactor

Detailed description of the reactor can be found elsewhere (Jonah et al., 2007). The reactor was modified to include two additional safety control rods totally inserted in the inner irradiation channels located at angles 0° and 144° as displayed in Fig. 1a–c. The inner irradiation channels are placed 40 mm above the bottom of the core through which the ASCRs move to cover a distance of 190 mm (i.e. 230–40 mm) in the core. The total vertical distance travelled by the ASCRs is 192 mm. The diameter of each ASCR is 18 mm including clad thickness of 0.5 mm. The geometric representation of the central control rod and ASCR when fully inserted as simulated by the MCNP is shown in Fig. 2a and b.

3. MCNP modeling and simulation

The existing input deck was modified with changes to the coolant room temperature of 27 °C and two inner irradiation channels with two ASCRs. Based on these operating conditions, most of the continuous energy neutron interaction data from the ENDF/B-VI cross section library for 300 K evaluation were used in the MCNP calculation. Where data were not available in ENDF/B-VI evaluation, the ENDF/B-V library data were used (Kinsey, 1979). Special $S(\alpha, \beta)$ treatment at 300 K for hydrogen in light water and beryllium were used to account for molecular binding effects below 4 eV (Stamatelatos et al., 2007) and to accurately simulate the reactor at room temperature.

In modeling the ASCR, the dimensions and material composition of the central control were optimized through the introduction an ASCR into one of the unconnected inner irradiation channels. The length of the cadmium was reduced in the case of the ASCR since the level of the irradiation channels are 40 mm above bottom core region. Criticality calculations were then performed with the ASCR in completely inserted position keeping the central control completely withdrawn until a sub-criticality was achieved. Runs performed for nuclear criticality did not give sub-criticality therefore; the absorber's surface area was increased. This was done up to a maximum diameter of 18 mm with 2 mm gap between the ASCR and the inner walls of the irradiation channel. Because sub-criticality was not achieved, another ASCR was introduced with the first kept in completely inserted position and the diameter was also increased gradually until sub-criticality was achieved. This was also achieved with a diameter of 18 mm for the second rod. The ASCRs were inserted into two unconnected inner irradiation channels located at angles 0° and 144° which were named A and B respectively. The choice to maintain properties of the main control rod was made because of ease of manufacture and good properties of cadmium absorber (Bretscher, 1997; Ismail, 2010; Shoushtari et al., 2010). Reactor safety rods must effectively absorb neutrons since they are mainly used for emergencies.

Preliminary calculations were performed to make the fission source converge from an initial guess distribution with arbitrary but uniform set of points in the fuel regions to estimate nuclear criticality, k_{eff} excess reactivity, ρ_{ex} and control rod worth, ρ_w using the KCODE option with rod(*s*) withdrawn and inserted as the case may be. In this work, the final runs for the KCODE involved typically 30 settle cycles followed by 800 cycles of 500,000 histories. Power iteration for Monte Carlo criticality calculation of the mean value of k_{eff} is shown in Fig. 3. Results obtained for each ASCR is presented in Table 1 for HEU and LEU cores. The ASCRs were also

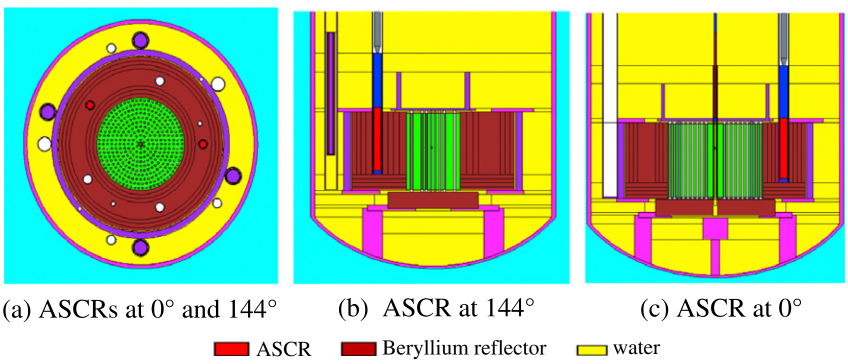

Fig. 1. Cross section through the NIRR-1 reactor showing ASCRs.

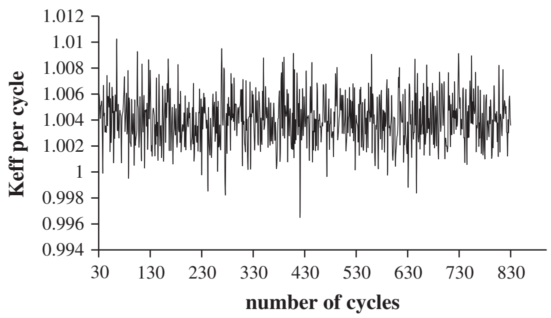

Fig. 3. Power iteration for Monte Carlo criticality calculation of the mean value of k_{eff}.

simulated while in completely inserted position to calculate the control rod worth and shutdown margin keeping the central control rod withdrawn.

The excess reactivity, ρ_{ex} was estimated by running the input with all rods withdrawn using the relation (Duderstadt and Hamilton, 1976).

$$\rho_{ex} = (k_{eff} - 1)/k_{eff} \tag{1}$$

where k_{eff} is the effective multiplication factor. The individual worth of each rod, ρ_w was obtained from the relation (Balogun, 2003).

$$\rho_w = \frac{k_{out} - k_{in}}{k_{out}} * k_{in} \tag{2}$$

where k_{in} and k_{out} are the effective multiplication factors for rod(s) inserted and withdrawn respectively. The shutdown margin (ρ_{sm}) for each rod was obtained from the relation (Balogun, 2003; Duderstadt and Hamilton, 1976).

$$\rho_w = \rho_{ex} + \rho_{sm} \tag{3}$$

The shutdown margin not only characterizes the core multiplication capability in its shutdown state, but is also related to the rate at which the power level may be reduced in an emergency shutdown. The fractional power level decrease achieved immediately after insertion of the ASCRs in the event of an emergency was calculated using the relation (Duderstadt and Hamilton, 1976).

$$\text{Power before control insertion}/\text{Power after control insertion} = \beta/(\rho sm + \beta) \tag{4}$$

where β is the effective delayed neutron fraction. The effective delayed neutron fraction was obtained from the relation (Klein Meulekamp and van der Marck, 2006).

$$\beta = 1 - k_p/k \tag{5}$$

where k is the effective multiplication factor for all neutrons and k_p is the effective multiplication factor for prompt neutrons, preventing the influence of the delayed neutrons and thereby obtaining the value of effective multiplication factor for prompt neutrons.To study the effect of the ASCRs on the neutron flux while withdrawn during reactor normal operation, track-length estimates of neutron flux (F4) tallies were used. The output was inserted in excel spreadsheet. The un-normalized particle flux tallies were then normalized to get the actual fluxes. The criticality calculation was normalized by the steady state power, P of the reactor using the expression:

$$P * (\text{number of fission neutrons produce}/\text{Watt-s}) * (\bar{U}) \tag{6}$$

where $\bar{U} = 1/(\text{loss to fission})$. The number of fissions neutrons produced was deduced from the MCNP output by checking for fission q-values for U-235 Isotope which is

$$\left(\frac{1\ \text{J/s}}{w}\right)\left(\frac{1\ \text{MeV}}{1.60205E - 13}\right)\left(\frac{\text{fission}}{180.88\ \text{MeV}}\right) = 3.450908E + 10\frac{\text{fission}}{W_s} \tag{7}$$

The loss to fission was also obtained from the MCNP output by searching for "loss to fission". For a steady state operation, the number of neutrons/fission (η) is approximately 2.4. The source strength is calculated by the factor

$$P(W) * 3.450908E + 10 * \bar{U} \tag{8}$$

The neutron flux tallies are normalized by using the normalization factor:

$$(P(W) * 3.450908E + 10 * \bar{U} * \text{tally})/\text{volume} \tag{9}$$

4. Results and discussion

Because the movement of the single control rod is controlled by mechanical clutches, it is possible for it to malfunction and therefore may not perform its intended function hence the need for additional control rods to improve safety. To circumvent this problem, two ASCRs were introduced to achieve sub-criticality with the central control rod withdrawn.

The power iteration for Monte Carlo criticality calculation of the mean value of k_{eff} is shown in Fig. 3. The result obtained for the nuclear criticality, excess reactivity and control rod worth for the HEU core were compared with measured value to check the effect of the modification of the cross section data at typical pool temperature. The result obtained for HEU and LEU cores are presented in Table 1.

Table 1 shows some eigenvalues with relative errors for NIRR-1 when rods are out of the core and also the eigenvalues when each of the three rods is fully inserted in the core for HEU and LEU. The worth of each rod was calculated. The worth of main control rod was found to be 7.62 mk for the HEU which is quite close to the measured value of 7.0 mk (SAR, 2005) and value reported by some authors (Jonah et al., 2007) while that of LEU was found to be

Table 1
Calculated neutronics parameters for NIRR-1 HEU and LEU cores.

Parameter	Experiment	HEU			LEU		
Rods-out (k_{eff})	–	1.00474 ± 00021			1.00474 ± 0.00021		
Main rod-in (k_{eff})	–	0.99714 ± 0.00021			0.99824 ± 0.00021		
A-in k_{eff}	–	1.00165 ± 0.00021			1.00164 ± 0.00021		
B-in k_{eff}	–	1.00169 ± 0.00021			1.00181 ± 0.00021		
Core excess reactivity ρex (mk)	4.99–1.2 = 3.77	4.72 ± 0.05			4.72 ± 0.05		
Worth of rods (mk)	7.0	Main	A	B	Main	A	B
		7.59	3.07	3.03	6.48	2.97	2.91
Total worths for ASCRs (mk)	–	6.10				5.88	
Shutdown margin (mk)							
Main CR	3.33		2.87			1.76	
A and B			1.38			1.16	
Effective delayed neutron (β_{eff})	0.0081		0.00834 ± 0.0008			0.00849 ± 0.0008	

Table 2
Calculated flux for NIRR-1 HEU and LEU inner and outer irradiation channels.

Fuel type	Thermal 0.625 eV($ncm^{-2}\,s^{-1}$) × 10^{12}		Epithermal 0.625 eV–0.825 MeV $ncm^{-2}\,s^{-1}$ × 10^{12}		Fast (0.825–20 MeV)$ncm^{-2}\,s^{-1}$ × 10^{12}	
	Inner	Outer	Inner	Outer	Inner	Outer
HEU	1.16 ± 0.01	0.66 ± 0.01	1.29 ± 0.01	0.19 ± 0.01	0.27 ± 0.01	0.04 ± 0.003
LEU	1.04 ± 0.01	0.62 ± 0.01	1.26 ± 0.01	1.18 ± 0.01	0.26 ± 0.01	0.04 ± 0.003

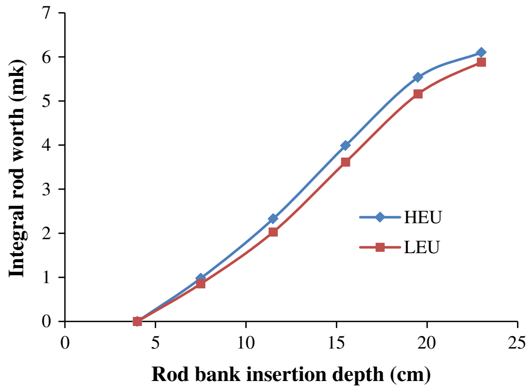

Fig. 4. Integral worth of ASCRs versus depth from the bottom of inner channels for HEU and LEU cores.

6.48 mk which is in agreement with reported value (Jonah et al., 2009). This decrease in rod efficiency is expected for the LEU and can be attributed to the increase in the inventory of ^{238}U in LEU resulting in the hardening of the spectrum. The worth of A and B was calculated to be 3.07 mk and 3.03 mk for HEU and 2.97 and 2.91 for LEU respectively. The combined worth for A and B were calculated to be 6.10 mk and 2.88 mk for HEU and LEU respectively. In both cores, sub-criticality was achieved. During the zero power experiment, the available excess reactivity for NIRR-1 was 4.97 mk. Due to licensing condition, a permanent poison of worth (−1.2 mk) to bring the value to 3.77 mk. The core excess reactivity of 4.72 ± 0.05 is quite close to the measured value.

In the estimation of the effective delayed neutron fraction, two calculations were made. In the first the parameter "totnu yes" and the second "totnu no" were used while rods A and B were withdrawn and the value of the effective delayed neutron fraction obtained are presented in Table 1 for HEU and LEU cores. The effective delayed neutron fraction obtained for the HEU and LEU cores are statistically indistinguishable. This suggests that effective delayed neutron fraction does not significantly depend on the fuel type. This was also observed by some authors (Snoj et al., 2010).

The shutdown margin of 2.87 mk obtained for the central control rod for HEU core is quite close to the measured value of 3.33 mk (SAR, 2005) and reported value (Jonah et al., 2007). This value is lower for the LEU as expected. The shutdown margin for the ASCRs for HEU and LEU cores were calculated to be 1.38 mk and 1.16 mk respectively. Again, the same is observed with the ASCRs as you move from HEU to LEU. Using this value calculated for ASCRs in Eq. (4) implies that in the event of rod stuck situation of the central control rod withdrawn or when the rods are scrammed into the reactor, the power level will drop by 90% and 88% for HEU and LEU cores respectively.

Because NIRR-1 is mainly designed for NAA, any modification should not significantly affect the neutron flux for routine experiments. The flux magnitudes calculated in the inner and outer channels for HEU and LEU cores are presented in Table 2.

From the results in Table 2, it can be seen that the calculated thermal neutron flux value in the inner channel for the HEU core was $(1.16 \pm 0.01) \times 10^{12}$ n/cm^2 s at an axial height of −3.32 cm. This value is in good agreement with the value of $(1.1 \pm 0.2) \times 10^{12}$ n/cm^2 s (SAR, 2005) measured during the steady state operation at full power (30 kW) at commissioning and reported value (Jonah et al., 2007). This value is also consistent with measured data of $(5.4 \pm 0.2) \times 10^{11}$ n/cm^2 s (Jonah et al., 2005) using Al–0.1%Au foil activation detector at a steady state power of 15 kW. For the LEU, the calculated value obtained was $(1.04 \pm 0.01) \times 10^{12}$ n/cm^2 s which agrees with reported value (Jonah et al., 2009). This suggests that there is a decrease in the flux as you move from HEU to LEU. Since the ASCRs will remain withdrawn during normal operation, their effect on flux distribution when inserted was not studied. As in accordance with calculations and measurements performed for the main control rod, the integral worths of ASCRs were calculated. Data obtained as the ASCRs are inserted in bank, a plot of reactivity versus depth of insertion for the HEU and LEU cores are shown in Fig. 4.

5. Conclusions

Because the movement of the single control rod is controlled by mechanical clutches, it is possible for the mechanical systems to malfunction and therefore may not allow the control rod to perform its intended function hence, the need for additional control rod. The MCNP model of NIRR-1 developed for the conversion of NIRR-1 to LEU was modified to simulate two ASCRs to enhance safety .Neutronics data obtained indicates that it is possible to include two ASCRs with little or no changes to the existing HEU core. Furthermore, the introduction of the two ASCRs has no effect on the neutron distribution in completely out position.

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