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Reactivity temperature coefficients for the HEU and LEU fuel of the MNSR reactor

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ABSTRACT

Analysis of the Reactivity Temperature Coefficients of the Miniature Neutron Source Reactor (MNSR) for normal and accidental conditions (above 45 °C) using HEU-UAl_4 and the LEU: U_3Si, U_3Si_2 and U9Mo fuel were carried out in this paper. The Fuel Temperature Coefficient (FTC), Moderator Temperature Coefficient (MTC), and Moderator Density Coefficient (MDC) were calculated using the GETERA code. The contribution of each isotope presented in the fuel cell was calculated for the temperature range of 20 °C –100 °C at the beginning of the core life. The average values of the FTC for the UAl_4, U_3Si, U_3Si_2 and U9Mo were found to be: −2.23E-03, −1.85E-02, −1.96E-02, −1.85E-02 mk/°C respectively. The average values of the MTC for the UAl_4, U_3Si, U_3Si_2 and U9Mo were observed to be: −8.91E-03, −1.24E-04, −4.70E-03, 2.10E-03 mk/°C respectively. Finally, the average values of the MDC for the UAl_4, U_3Si, U_3Si_2 and U9Mo were observed to be: −2.06E-01, −2.03E-01, −2.04E-01, −2.03E-01 mk/°C respectively. It's found also that the dominant reactivity coefficient for all types of fuel is the MDC.

1. Introduction

The MNSR Research Reactor, with ≈ 90% enriched HEU- UAl_4 fuel is considered to be an inherently safe reactor since it is designed with a large negative reactivity feedback (CIAE, 1993). It has a fuel burn up of 1% and designed for a 10 year core lifetime before the next core replacement if the reactor is operated at its maximum flux for 2.5 h a day, five days a week (Khamis and Khattab, 2000). Due to the small size of the core which facilitates neutron leakage, the core is heavily reflected on the side and underneath the fuel cage by a thick annulus and slab of beryllium alloy material. This is done to minimize the neutron leakage and hence conserve the neutron economy in the reactor core. Due to the burn up of the fuel in the core, regulatory shims of beryllium have been added to the top aluminum tray to compensate for the loss of the reactivity resulting from the fuel burn up and accumulation of fission products. Under normal operating conditions the core excess reactivity decreases from ≈ 4mk due to fuel burn-up. Therefore, the future cycle operations will require a replacement of the spent or depleted fuel with a fresh fuel core which could either be HEU or LEU fuels.

Since 2002, international efforts to convert research reactors that are fuelled with HEU to LEU fuels have been intensified significantly through the United States Department of Energy's Reduced Enrichment for Research and Test Reactors (US DOE RERTR) program (Goldman and Adelfang, 2007).

LEU fuels such as: U_3Si, U_3Si_2 and U9Mo have been recommended to be used in the next nuclear fuel cycle operation for the MNSR (Matos and Lell, 2005). But this effort presents several challenges. Due to the increased amount of the U-238 in the LEU fuel, greater absorptions of neutrons due to temperature increase (Doppler broadening) in the resonance region is expected as fast neutrons slow down to thermal energies. This increase in resonance neutron capture affects criticality and hence core reactivity which has associated nuclear safety challenges and therefore must be investigated. The reactivity temperature coefficient (RTC) is of interest in reactor design for assessing the magnitude of any reactivity drift caused by temperature changes during reactor operation and also the reactor inherent safety features (Wajima and Yamamoto, 1965).

The work presented in this paper evaluated the fuel temperature coefficient, moderator temperature coefficient and moderator density coefficient for the MNSR using the reference HEU-UAl_4 fuel and the proposed LEU fuels.

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2. MNSR research reactor

The MNSR was manufactured by the Chinese Institute of Atomic Energy (CIAE, 1993). It is a low power tank-in pool type research reactor. The reactor employs highly enriched uranium as fuel, light water as moderator and coolant, and metal beryllium as reflector. Heat generated in the core is removed through natural convection. There are 10 irradiation sites in the reactor. Five of them are inside the annular beryllium reflector and the other five surround the annular reflector externally. The maximum thermal neutron fluxes in the inner and the outer sites are 1.0×10^{12} and 5.0×10^{11} n/cm^2.s, respectively. The nominal thermal power of the reactor at this neutron flux level is 30 kW. The cold excess reactivity was adjusted to less than 4 mk using the reactivity regulators (see Fig. 1).

Inherent safety features of the MNSR are the availability of highly negative moderator temperature coefficient of reactivity (approximately −0.l mk/°C at a temperature range of 20–45 °C), and low critical mass of the core (CIAE, 1993). These characteristics limit achievable peak power levels following an accidental insertion of reactivity and assure the safety of the reactor under all conceivable accident conditions. Operating performance of the MNSR is limited to its inherent characteristics. Being under moderated, the MNSR core is very sensitive for any changes in moderator temperature. It is worthy mentioning that temperature effect due to fuel temperature increase is rather insignificant in the operating ranges of the MNSR, due to the fact that the highly enriched uranium is used as fuel i.e. little amount of ^{238}U which is a resonance absorber.

3. Method

3.1. The super unit cell model

The lattice structure of the MNSR core was represented in the form of a super unit cell which conserves the volumes of core materials, i.e. fuel, clad, moderator, dummy elements, and structure as can be seen in Fig. 2. The reflective boundary condition was used in the unit cell calculation using the GETERA code (Belousov et al., 1992). The GETERA code uses the multi-energy group discrete ordinate and collision probability methods for the unit cell calculation. The density numbers, temperature, dimension and compositions of the MNSR unit cell were given as input to the GETERA code which was used in calculations (Khattab and Dawahra, 2011). Table 1 presents the different fuel types as candidates for the MNSR LEU fuel which were used in this paper.

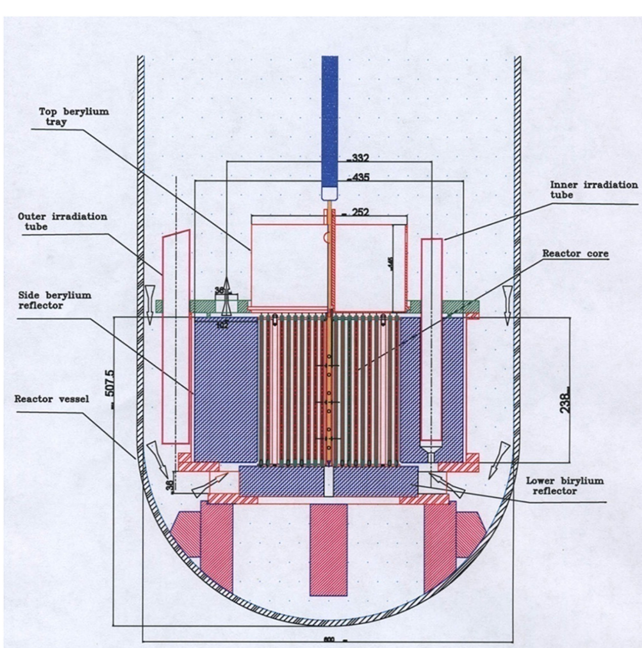

Fig. 1. Side view of the MNSR reactor.

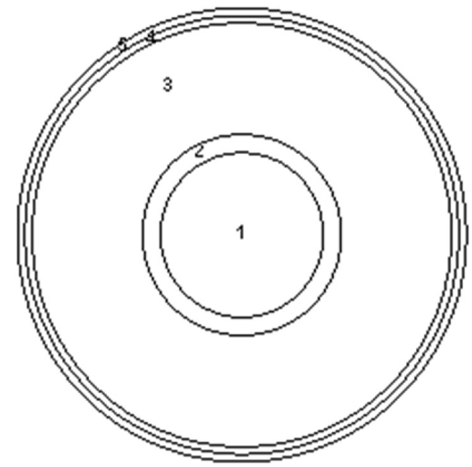

Fig. 2. The MNSR super unit cell was modeled by the GETERA code, where: 1- fuel (r = 0.215 cm), 2- clad (r = 0.275 cm), 3- moderator (r = 0.6182 cm), 4- dummy elements (r = 0.6187 cm), 5- structure (r = 0.6197 cm).

3.2. Buckling calculation of the reactor

The axial buckling was calculated using the following formula: $(\pi/H)^2 = (\pi/23)^2 = 0.018638$ cm^{-2}, where H is the reactor height. The GETERA code can calculate the radial buckling of the MNSR reactor using the CEFZAD card if the effective multiplication factor (k_{eff}) of the reactor is introduced as an input to the code. The k_{eff} of the MNSR was calculated previously and found to be 1.00517 at 20 °C (Khattab and Sulieman, 2009).

The reactivity temperature coefficient (α_T) is defined as the change in the reactivity (ρ) with respect to temperature as in the following equation:

$$\alpha_T = \frac{d\rho}{dT} \tag{1}$$

where, ρ can be calculated as follows:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \tag{2}$$

Differentiating Eq. (2) with respect to temperature (T) gives:

$$\alpha_T = \frac{d\rho}{dT} = \frac{d}{dT}\left(\frac{k_{eff} - 1}{k_{eff}}\right) = \frac{1}{k_{eff}^2}\frac{dk_{eff}}{dT} \approx \frac{1}{k_{eff}}\frac{dk_{eff}}{dT} \approx \frac{1}{k_{eff}}\frac{\Delta k_{eff}}{\Delta T} \tag{3}$$

This equation is valid assuming a liner relationship between the k_{eff} and the temperature in a small temperature range. The FTC was calculated by varying the fuel temperature and keeping the temperature and density of the moderator constant. The MDC was calculated by varying the moderator density and keeping the temperature of the moderator and fuel constant. The total MTC (sum of the temperature effect and the water density effect) was calculated to be compared with the previously published results as