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Investigation of fuel lattice pitch changes influence on reactor performance through evaluate the neutronic parameters

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ABSTRACT

Nuclear reactor core design is one of the most complex issues that nuclear engineers deal with. The number and complexity of effective parameters and their impact on reactor design, which makes the problem difficult to solve, require precise knowledge of these parameters and their influence on the reactor operation.

Numerous factors in a nuclear reactor core design depend on the Fuel-to-Moderator volume ratio, V_F/V_M, in a fuel cell. This ratio can be modified by changing the lattice pitch which is the thickness of water channels between fuels plates while keeping fuel slab dimensions fixed. Cooling and moderating properties of water are affected by such a change in a reactor core, and hence some parameters related to these properties might be changed.

The aim of this research is to provide the suitable knowledge for nuclear core designing. To reach this goal, the first operating core of Tehran Research Reactor (TRR) with different lattice pitches is simulated, and the effect of different lattice pitches on some parameters such as effective multiplication factor (K_{eff}), reactor life time, distribution of neutron flux and power density in the core, as well as moderator temperature and density coefficient of reactivity are evaluated. The nuclear reactor analysis code, MTR-PC package is employed to carry out the considered calculation. Finally, the results are presented in some tables and graphs that provide useful information for nuclear engineers in the nuclear reactor core design.

1. Introduction

The primary responsibility for the nuclear design of a reactor core rests with the nuclear engineer. He must determine that set of system parameters which will yield safe, reliable, and economical reactor operation at the rate power level over the desired core lifetime.The nuclear analysis of the core cannot be performed in an independent manner, but rather it must interact strongly with other aspects of core design, including thermal–hydraulic analysis of core cooling, structural analysis of core components, economic performance, and so on (Duderstadt and Hamilton, 1976).

The primary concern in the design of the reactor core is its nuclear analysis that defines the principle design activities of the nuclear engineer. The calculation of the core multiplication and flux or power distribution is, of course, the most common type of analysis performed in nuclear core studies. In addition, reactivity and control analysis must be performed to reach a flexible and safe reactor operation. From above discussion, it is apparent that the responsibilities of the nuclear designer are quite varied and numerous. He must establish the limitations and determine the value of core enrichment, moderator-to-fuel volume ratio, the fuel rod diameter or thickness of fuel plate, fuel element arrangement, core geometry and the location and type of reactivity control.

In order to optimize fuel utilization at Tehran Research Reactor (TRR), the method of fuel management is modified by using calculation code system and the result show that the core lifetime and average extracted burn up of spent fuel element of TRR are improved significantly (Keyvani et al., 2010). In addition, to improve the issues of control and reactivity of TRR core, neutronic calculation of current low enriched uranium control fuel element replacement with the high enriched uranium control fuel element in the reference core of TRR has been investigated (Ahmad Lashkari et al., 2012).

The fuel lattice pitch or Fuel-to-Moderator volume ratio is an important parameter in the reactor core design process. TRR is a light water moderated with the thermal power of 5 MW. The fuel element used in this reactor has fuel lattice pitch 0.42 cm (AEOI,

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2001). In this research, 13 lattice pitches are proposed for fuel elements. For all proposed lattice pitches, the fuel slab dimensions are fixed and just the thickness of water channels between fuel plates is varied, as a result the volume of fuel elements and so the core size changed. TRR with different lattice pitches is simulated utilizing the MTR-PC package, and based on the analysis of the core performance with different fuel lattice pitches, calculation of different parameters, as effective multiplication factor (K_{eff}), reactivity, coolant temperature and density coefficient of reactivity and distribution of multi-group neutron flux and power density at the core, for the first operating core of TRR is performed.

2. Fuel lattice pitch and its influence on neutronic parameters

In the reactor core with slab geometry, the distance between two consecutive fuel plates, is a symbol of the fuel lattice pitch. In a more detailed definition, the Fuel-to-Moderator volume ratio, V_F/V_M, in a fuel cell specifies the fuel lattice pitch. Due to the relation with two essential elements of the core, fuel and moderator, pitch is known as one of the important design criteria of the reactor core. By changing the volume of fuel or moderator or both of them the fuel lattice pitch is varied and affects on important neutronic parameters. In this research the lattice pitch is changed by keeping the fuel slab dimensions fixed, and only the thickness of water channel is changed, i.e. the volume of fuel is fixed and the volume of moderator is varied. Variation of moderator volume has the greatest impact on two basic parameters, the utilization factor, f, and resonance escape probability, p (Lamarsh, 1966). These are two factors of the six-factor formula (Eq. (1)) (Duderstadt and Hamilton, 1976). For example by decreasing the moderator volume, the neutron absorption to moderator (non-fuel material) is reduced and according to Eqs. (1) and (2), utilization factor and multiplication factor are increased. On the other hand, decreasing the amount of the moderator may cause a change in the neutron spectrum; this reduction is associated with decline of resonance escape probability, p, and according to Eq. (2) can reduce the multiplication factor. So there is a competition between f and p; by change to moderator volume, one of them has a positive effect on the multiplication factor and the other one has negative effect. In some pitches positive effect is dominant and in some ones the negative effect.

$$f = \frac{\text{neutron absorption to fuel}}{\text{neutron absorption to non-fuel}}$$

$$f = \frac{\Sigma_{\alpha F} V_F \varphi_F}{\Sigma_{\alpha F} V_F \varphi_F + \Sigma_{\alpha M} V_M \varphi_M} \tag{1}$$

$$k = \varepsilon \eta f p P_{TN} P_{FN} \tag{2}$$

where

η: average number of neutrons produced per neutron absorbed in fuel.
f: utilization factor.
p: resonance escape probability.
P_{FNL}: fast neutron non-leakage.
P_{TNL}: thermal neutron non-leakage.

Moreover, changes in the fuel lattice pitch affect the reactivity coefficients. Eq. (3) shows the relationship between lattice pitch and moderator temperature coefficient (Driscoll et al., 1990).

Moderator Temperature Coefficient:

$$\mathrm{MTC} = \frac{\partial \rho}{\partial T_M} = -\alpha \left(\frac{V_F}{V_M}\right) \frac{\partial \rho}{\partial \left(\frac{V_F}{V_M}\right)} \tag{3}$$

$\alpha = 1/V_M(\partial V_M/\partial V_F)$, the volumetric expansion coefficient for H_2O.

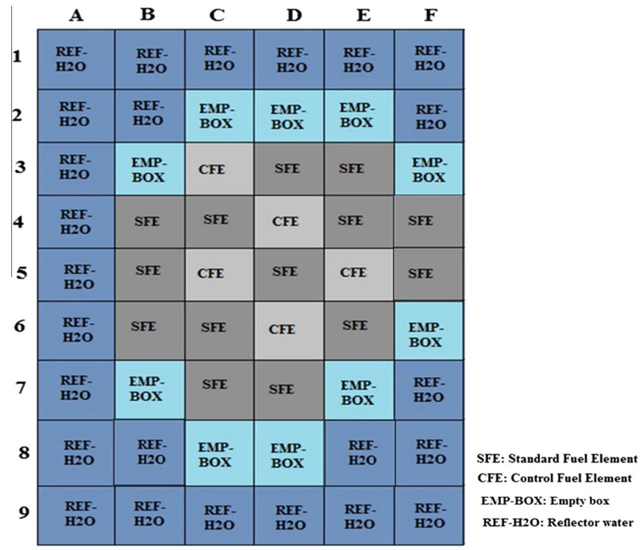

Fig. 1. TRR first operating core configuration.

3. Description of TRR

In this research TRR reactor selected to perform the analysis. TRR is an open pool type, light water moderated with the thermal power of 5 MW (AEOI, 2001). Its core configuration contains MTR-fuel type elements of low enriched uranium (LEU) with 20% enrichment that are arranged in 9×6 grid plate assembly. The core configuration of the first operating core which allows reactor operation at maximum power level (5 MW) is shown in Fig. 1. This core configuration is water reflected, and has 14 Standard Fuel Elements (SFE), 5 Control Fuel Elements (CFE) and 10 empty boxes. The general layout of SFE and CFE are given in Fig. 2(a) and (b) (AEOI, 1989). Benchmark specifications of fuel elements of TRR are explained in Table 1.

4. Materials

The MTR-PC package has been developed by INVAP S.E in order to perform neutronic, thermal hydraulic and shielding calculations of MTR-type reactor for personal computers (PC). In neutronic calculation, the methodology is divided in three steps. These steps are summarized in a calculation line with the following items: working library generation, cell calculation and core calculation.

Working library generation: The INVAP WIMS library has updated from ENDF/BIV.

Cell calculation: This calculation step is performed with the transport code WIMS, to generate a homogenized macroscopic cross section set for each component of the core. These macroscopic cross sections are used as input for the next step.

Core calculation: This calculation step is performed with the diffusion code CITVAP, and outputs are needed neutronic parameters. Fig. 3 shows this neutronic calculation line scheme.

The WIMS code is used to calculate fuel cell or fuel-rod cluster, as well as fuel plate with slab geometry. For MTR-type reactors, calculations are conducted in one dimensional geometry (1D), applying the collision probability method. WIMS code is utilized for macroscopic cross-section generation (Lawrence and Taubman, 1980). POS-WIMS pose-processor condenses and homogenizes macroscopic cross-section from WIMS output.

The CITVAP code is an improved version of the well known CITATION II code. It solves 1, 2 or 3-dimensional multi-group diffusion equations in rectangular or cylindrical geometry. Nuclear data