OPTIMIZATION OF MATERIAL TEST RESEARCH REACTOR CORE FOR ISOTOPE PRODUCTION

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Abstract

In this paper Neutronic calculation (deterministic approach) has been carried out to characterize the neutron flux in locations used for isotope production to achieve optimum utilization of the reactor. In neutronic calculations a number of approximations take place. As a result even if one uses the same codes the results might be different. It is thus of importance to care about the model used. Comparison between different model and with published experimental results has been done. In this work three deferent model of the standard fuel has been tested to determine the accurate one, then it used for core calculation. Calculation of reactivities at deferent cycles, calculation of power densities, neutron flux, and burn-up as well as search for equilibrium were performed to determine the equilibrium core. Standard computer codes WIMSD-5B and CITVAP were used.

1. INTRODUCTION

As general currently operational research reactors can be categorized into pool-type material test reactors (MTRs), Triga-type, Tank-type, homogeneous-type and miniature neutron source reactors (MNSR). The comparison of decommissioned and shutdown reactors shows dominance of open-pool type MTRs. Also, according to the latest statistics two out of eight under construction research reactors are MTRs. These reactors generally have beam-tubes, thermal columns as well as in core irradiation facilities while easy access to the reactor core enables one to perform a large variety of experiments. The MTRs type some time discussed as Multi-purpose research reactor and high flux research reactors. High flux research reactors play a crucial role in providing unique facilities for basic research in frontier areas of science and for applied research related to development and testing of nuclear fuels and other materials. These reactors also cater to the increasing need of radioisotopes for application in the field of medicine, agriculture and industry.

Reactor core neutronic calculations are one of the important analyses that help the best utilization plans and the optimum economical parameters. It is necessary to characterize the neutron flux for every facility in the reactor to achieve optimum utilization of the reactor. Also, it is necessary to characterize the neutron flux in positions used for radioisotope production in the reactor to evaluate the radioisotopes yields. The resulted flux data will be a guide for radioisotope production program. The utilization capacity of a research reactor in terms of radio isotope production is directly related to the magnitude of the neutron flux and to the nature of the neutron spectra present in the irradiation positions. The neutron flux can be achieved experimentally by measurement of neutron flux, or theoretically by neutronic calculations.

1.1. PARR-1 description

PARR-1 is typical MTR type, initially designed to operate on 93% HEU fuel at 5 MW. Now it has been converted to 20% LEU fuel with upgrading of the power to 10 MW. The LEU core of it consists of standard and control fuel elements. The MTR type fuel element used is U_3Si_2-Al and the enrichment is 19.99% U-235.

2. NEUTRONIC CALCULATION

Cell level calculation had been done using WIMS code with the original library. And for core level calculation CITVAP code and local subroutines had been used.

According to an earlier work, search for equilibrium core could take place by three different methods. These methods use different criteria for the EOC during the iterative procedure of finding equilibrium core. The three methods are EOC reactivity, Fixed Cycle Length, and Maximum Allowed Discharge Burn-up.

In the present work, Fixed Cycle Length method had been used for equilibrium core search, in the beginning the core with all fresh fuel elements is burnt in cycles with equal length of 40 days. The reshuffling had been done according to the recommended reshuffling scheme. The equilibrium core had been achieved after 6 cycles.

3. RESULT & ANALYSIS

Three models of the standard fuel element was tested using WIMS code, we found the good one is to model the extra part of the fuel plate and the side aluminium plate as a separated region then homogenised all. We refer this to that; by adding these parts to the fuel plate we could change the fuel to moderator ratio.

3.1. Core calculation

When the core was initially burnt with equal cycle length of 40 days, the reactivity at BOC and EOC decreases exponentially with cycle number and saturates at certain value after 200 days as shown in Figure 2.

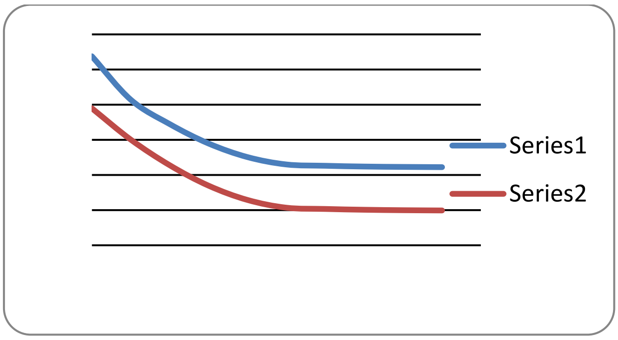

FIG. 2 Reactivity of the core at BOC and EOC with fixed cycle length

Figure 2 shows the values of BOC and EOC reactivity by series1 and series2 respectively. The core begins to saturate at cycle No. 5, for accurate saturation estimation cycle No. 6 is had taken as equilibrium core. Thus in it we can study different neutronic parameters like neutron flux, power density, that help in the determination of the amount of fission products in the deferent fuel elements and deferent burn-up.

TABLE 1. COMPARISON BETWEEN CALCULATED, MEASURED, AND RESULT OF REF(EXPERIMENTAL PUBLISHED VALUE):

Core parameters	Calculated	Ref( )	Measured
Fuel cycle length	40	40	40
Multiplication factor at EOC	1.00068	1.01	1.01
Neutron flux at the central flux trap n/cm^2.s	1.05×10^{14}	1.62×10^{14}	$1.5\text{x}10^{14}$

4. CONCLUSION

The result for reactivity and flux at BOC and EOC when searching for equilibrium core had been near good agreement with the experimental result which mean that the model used for core calculation is valid to be used for details investigation of neutronic parameters and safety analysis. But more investigations will be done to determine the reason for difference in multiplication factor at EOC

The continuation of this study deal with the determination of the amount of fission products and safety parameters for this core while try to get high level of flux in some experimental location through testing new reshuffling scheme, besides testing other types of fuel and different density loading .

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