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Depletion Calculations for MTR Core Using MCNPX and Multi-Group Nodal Diffusion Methods

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1. Introduction

During the operation of a nuclear system, the nuclide concentration will change as isotopes consume radiation and undergo various nuclear reactions. The temporal change in isotope concentrations results in further changes in scatter, capture, and fission events as isotopes either transmutate or fission into new nuclides possessing different probabilities for these reactions. For example, as U-235 depletes as a result of the fission process, fission products are generated. Because these newly created fission products possess capture reaction probabilities that rival U-235 for neutron capture, the fission products, combined with the depletion of U-235, reduce the propensity of the system to fission and maintain a steady-state chain reaction. In order to maintain a self-sustaining steady-state chain reaction, more fuel than is necessary in order to maintain a steady state chain reaction must be loaded. The introduction of this excess fuel increases the net multiplication capability of the system.

In this paper MCNPX and multi-group nodal diffusion theory will be used for depletion calculations for MTR core. The eigenvalue and power distribution in the core will be compared for different burnup.

2. Method and Model

2.1 MCNPX

The MCNPX is a general-purpose Monte Carlo N-Particle code that has been developed as an extension of the MCNP code. CINDER90 depletion code was integrated into the latest version of MCNPX to provide it with built-in burnup capabilities. CINDER 90 has inherent decay and 63-group cross-section data library for 3400 isotopes. In general, CINDER 90 utilizes one group cross sections tallied by MCNPX. However, for isotopes without continuous energy cross sections data, one group cross sections are obtained by collapsing 63-group CINDER90 cross-section set and 63-group neutron spectrum calculated by MCNPX. MCNPX utilizes predictor–corrector procedure.

2.2 NEWT-TRITON System

NEWT is a multi-group discrete-ordinates transport code with flexible meshing capabilities that allow 2D neutron transport calculations using complex geometric models. NEWT can be used to prepare a collapsed weighted cross section. The primary function of NEWT is to calculate the spatial flux distributions within a nuclear system and collapse the cross sections into multiple energy groups as specified by the user. It is used as a part of the TRITON depletion sequence, NEWT provides spatial fluxes, weighted multi-group cross sections, and power distributions used for multidimensional depletion calculations.

2.3 Multi-group nodal diffusion code

A multi-group 3D nodal diffusion code was developed to perform reactor calculation. This code uses unified nodal method (UNM) and it was validated using some benchmark problems. To perform the nodal calculations homogenized cross sections for each node must be obtained, for this purpose NEWT code was used to generate homogenized cross sections for the nodal code

2.4 Model

The reference model used in this framework for MTR core is the following:

1. A 3x3 MTR core with vacuum conditions in the radial and axial direction, in order to simulate a typical MTR core.
2. The considered reactor core has 9 fuel assemblies, each fuel assembly contain 21 fuel plates with an average power of 3 MW and an average power density that is around 156.96 MW/MTU.
3. Fuel material is U-7Mo and the fuel temperature and other materials 293 K, fuel composition is shown in table 1.
4. Beryllium depletion is not considered for this case.

Table 1. Isotopic composition of U-7Mo fuel

Isotope	wt%
U-234	1.090E-01
U-235	1.362E+01
U-236	1.519E-01
U-238	5.510E+01
Mo	5.203E+00
Al	2.583E+01
Total	100
U density	5.0 gU/cc
Fuel Density	7.234 g/cc
U:Mo	93:7

The assumed MTR core consists of 9 fuel assemblies of plate type fuel in 3x3 arrangements. Each assembly contains 21 fuel plates. It is reflected with beryllium at all sides except at top and bottom, and moderated with light water. For MCNPX the model describes the fuel assemblies by individual fuel plates. Each beryllium reflector element is modeled separately. The core is shown in Figure (1).

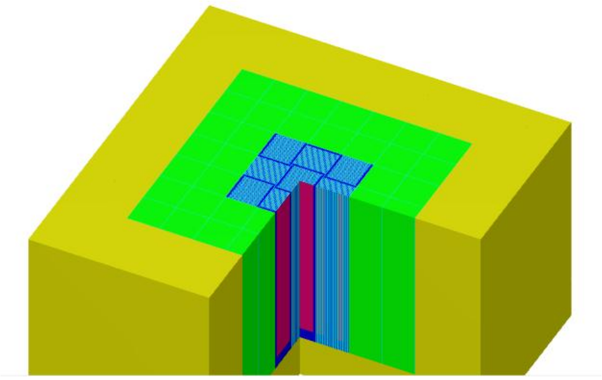

Fig.1. A 3D view of the 3x3 core

3. Results

The modeled MTR core was used to perform depletion calculations using MCNPX 2.7. The calculations performed by MCNPX were also performed by 3D nodal code for the assumed core using 2-groups and 6-groups cross sections generated by NEWT-TRITON system. In as far as possible the same or at least a similar approach has been followed. For example, in the set of calculations where all fuel assemblies in the core have the same burn-up which is equal to the average burnup in MCNPX. The results for k_{eff} with percent U-235 depletion is shown if figure (2).

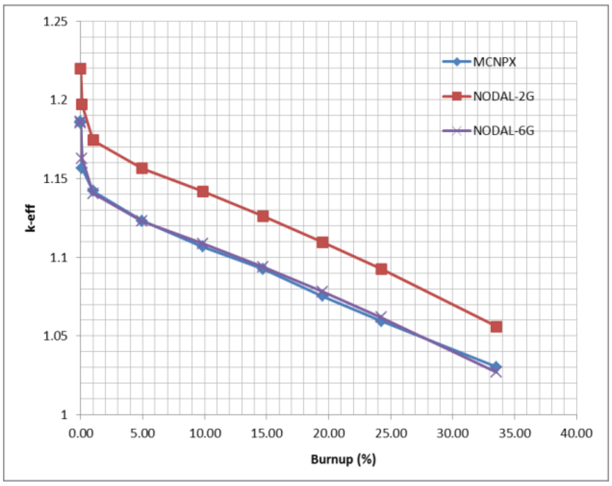

Fig.2. k_{eff} with percent U-235 depletion

The calculated U-235 percent burnup using MCNPX is shown in figure 3, also the average burnup for the core at different time steps 10, 100, and 350 days are given in figure 3. This will helps to know the value of U-235 burnup at which the cross sections should be generated in order to compare with MCNPX results. In figure 4 the calculated power for fresh and 33% burnup core using MCNPX and 6-groups nodal calculations are given.

1.03% 10.14% 33.80%	0.99% 9.83% 33.24%	1.03% 10.06% 34.07%
0.99% 9.79% 13.75%	0.91% 10 days 9.23% 100 days 14.42% 350 days	0.99% 9.63% 0.00%
1.03% 10.10% 34.55%	0.99% 9.79% 33.44%	0.99% 10.06% 34.39%

Avg. Bu 10 days 0.99 100 days 9.85
350 days 33.55

Fig. 3. Fuel assembly U-235 percent burnup from MCNPX.

Fresh core

0.3440 0.345066	0.3292 0.328177	0.3502 0.345066
0.3234 0.328177	0.3163 MCNPX 0.30703 NODAL	0.3332 0.328177
0.3321 0.345066	0.3277 0.328177	0.3440 0.345066

33% U-235 Burnup

0.3446 0.344351	0.3410 0.328629	0.3427 0.344351
0.3321 0.328629	0.3131 MCNPX 0.308081 NODAL	0.3273 0.328629
0.3429 0.344351	0.3280 0.328629	0.3282 0.344351

Fig. 4. Power of each fuel Assembly

4. Conclusions

Multi-group nodal diffusion theory with combination of NEWT-TRITON system was used to perform depletion calculations for 3x3 MTR core. 2G and 6G approximations were used and compared with MCNPX results for 2G approximation the maximum difference from MCNPX was 40 mk and for 6G approximation was 6 mk which is comparable to the MCNPX results. The calculated power using nodal code was almost the same MCNPX results. Finally the results of the multi-group nodal theory were acceptable and comparable to the calculated using MCNPX.

ACKNOWLEDGMENT

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