

**22.251 Systems Analysis of the Nuclear Fuel Cycle**  
**Fall 2009**  
**HOMEWORK SET #6**

**Problem 1. In problem set 2, you calculated the U236 concentration in recycle uranium that is re-enriched to 5 w/o. Suppose that this re-enriched recycle uranium is used to make a batch of fresh, 5 w/o fuel.**

**Use a two group method to evaluate how the U236 affects  $k_{\infty}$  starting with the following data from a CASMO calc for 5 w/o, 17X17, PWR fuel with no U236:**

* TWO GROUP DATA	K-INF, NO XE 1.39751	
DIFF1 , DIFF2	1.4360E+00	3.5052E-01
ABS1 , ABS2	1.0953E-02	1.1879E-01
NUFISS1 , NUFISS2	9.6800E-03	2.1653E-01
REMOV1 , NU	1.3229E-02	2.4451E+00
KAPPA , XE-YIELD	3.2456E-11	6.5790E-02

1.1 Calculate  $k_{\infty}$  without U236 from the two group macros using the formula:

$$k_{\infty} = \frac{\nu \Sigma_{f1}}{(\Sigma_{a1} + \Sigma_{R1})} + \frac{\Sigma_{R1}}{(\Sigma_{a1} + \Sigma_{R1})} \frac{\nu \Sigma_{f2}}{\Sigma_{a2}}$$

and verify that it agrees with the value given above. The macros are homogenized over the whole assembly. Group 1 is the fast group and group 2 is the thermal group. R1 refers to REMOV1, the downscattering macro from group 1 to group 2. There is no upscattering from group 2 to group 1.

1.2 Calculate the average number density of U235 in the assembly assuming its dimensions are those given in the input to the CASMO assembly problem in the Lab.

1.3 What is the number density of U236 in the recycle fuel assembly?

- 1.4 The U236 capture micro is 7.2 barns in group 1 and 3.1 barns in group 2. Use these to calculate the U236 macro cross sections in groups 1 and 2.
- 1.5 Add the U236 macros to the absorption macros for the assembly to calculate the new  $k_{\infty}$  with the U236 poison, and calculate the U236 reactivity worth.
- 1.6 How much must you increase the U235 enrichment to achieve the original  $k_{\infty}$ ? (Assume all the fission is due to U235.)

**Problem 2 LRM (Linear Reactivity Model) analysis.** Calculate and plot the LRM line (enrichment vs core average burnup) for the data in Slide 9 of Berger, H. D. "Advances in Reactor Core Fuel Management." September 16, 2008. Include the three numbered points plus a couple of points for very low and very high burnups. Do they all fall on to a single line?

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