Analysis of VHTR's with the SCALE System

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INTRODUCTION

This communication describes the use of the SCALE system[1] to design and analyze Very High Temperature Reactors (VHTR) intended for dual use: production of energy and the burning of Pu. One of the characteristics of this kind of reactor is the double heterogeneity effects Burnup calculations at assembly levels using multigroup transport theory imply three transport calculations: 1) one related to the spherical cell associate to the fuel kernels and their layers and the graphite matrix in between; at this step there is a cell homogenization to calculate multigroup space-average cross sections with Dancoff Factors for the resonance region, 2) a second cell calculation ,either describing one pebble or one cylindrical cell in the prismatic assembly, is used to calculate multigroup space-average cross sections, this time with a considerable increase of the size of the spatial average, and finally 3) with the two-times averaged cross sections from steps 1 and 2, the assembly is calculated; cross section are further spatial and energy averaged, this time to few groups (three) in order to compute the fuel depletion.

Because the SCALE module SAS2 performs two, rather the three, transport calculations per depletion step we decided to mimic the SAS2 sequence explicitly with the modules BONAMI, NITWAL, XSDRNPM, COUPLE, ORIGEN-S of the SCALE system, and of course include the additional transport calculation. One crucial step in this process is the calculations of Dancoff Factors. The factors used in this work were calculated with the Monte Carlo code MCNP.[2] We present the case of prismatic fuels with Pu-Np oxide fuel; the isotopic composition corresponds to discharged LWR fuel rods. We compare results of assembly eigenvalue calculations made with MCNP and the SCALE system (with proper Dancoff Factors).

The reference prismatic VHTR fuel assembly used for this analysis is shown in Figure 1.

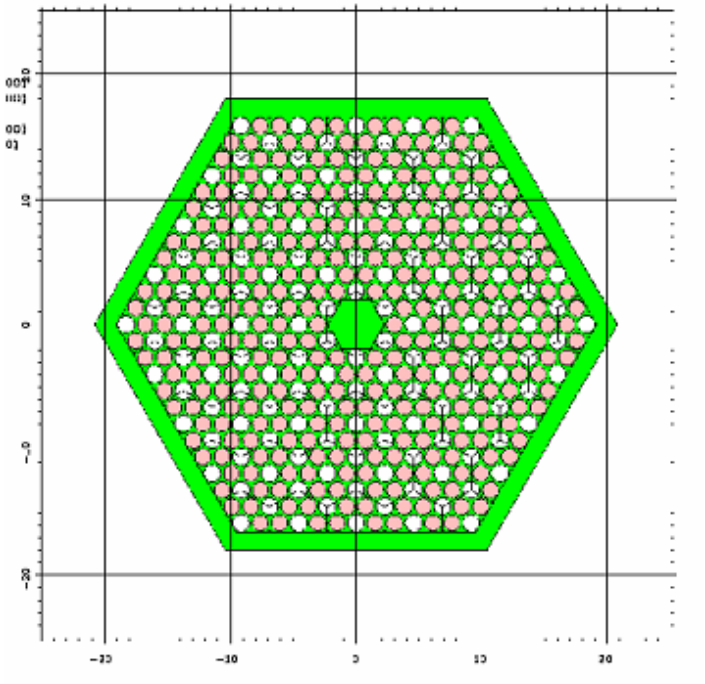

Figure 1: Cross section of the prismatic fuel assembly

CALCULATIONS OF THE DANCOFF COEFFICIENT

Dancoff Factors are used in the first heterogeneous calculations. Because of the finite nature of the kernel distributions: 3-D finite for pebbles and 2-D finite for prismatic fuels, resonance energy neutrons leaving one kernel can have an interaction not only inside the cluster where it is located but also in clusters that belong to other pebbles or compacts. Therefore the Dancoff coefficient for the kernels is space dependent. The holes for the fuel compact and the coolant have radii of 0.635 cm and 0.684 cm respectively. They are located in a hexagonal lattice (Fig. 1) of side 2.64 cm where there is, on average, one coolant hole per two fuel rods. The flat-to-flat distance of the assembly is 33.05 cm. Dancoff Factors were calculated for kernels at the center of the fuel compact for a central rod in the assembly of Figure 1; they might be different for the case of rods near the periphery. The calculations were done for various combinations of kernels radii and kernel concentrations in such a way that their combinations keep constant the Pu density in the compact (such that the atomic ratio n(C)/n(Pu)=4000). We have used white boundary conditions at the periphery of the assembly, keeping track of the resonance neutrons that leave the boundaries without interacting with the media. For the case of a central kernel in a central rod this current is very small so the results are independent of what is located beyond the assembly. The four layers, buffer, Py-C, CSi and Py-C, around the fuel kernels were explicitly modeled. Table 1 shows the results for the simple-cubic and body-centered-cubic lattice arrangements of the kernels used to mockup their real random distribution and the influence on the Dancoff factors. All the calculations were done with the body-centered-cubic lattice because it is more isotropic (or more "random") than the simple cubic lattice.

ANALYSIS OF PU BURNING IN VHTR's WITH THE SCALE SYSTEM

We made an extensive check of the capabilities of the SCALE system to analyze the burning of Pu in VHTR's. For this purpose the appropriate Dancoff Factors were computed according to the prescriptions of the previous section. The case is labeled "1 gram of Pu per pebble" because all the cases of Table 1 would produce that density for a standard pebble. The moderation ratio is approximately equal to 4,000. Whenever possible comparisons were made with MCNP calculations.

System at BOL

The beginning of life (BOL) case was compared with MCNP directly. As we discussed in the introduction the steps in the sequence of the SCALE calculations involve transport calculations of the assembly. We used a 1-D version of a real 2-D system, something that might be improved by using true 2-D transport analysis with the NEWT code of SCALE. In addition SCALE calculations are based on ENDF-B/V rather than ENDF-B/VI. Therefore the comparisons of the results with the MCNP calculations have to consider all these differences. For a system build with 100 microns radius fuel kernels (Case 1 in Table 1) k(MCNP)=1.288(2) and k(SCALE)=1.282.

Burnup Analysis with SCALE

The SCALE module COUPLE updates the cross sections of the standard libraries of ORIGEN-S, with actinides and fission product cross sections averaged in space and energy with the assembly XSDRNPM transport calculations. ORIGEN-S calculates the number densities for intermediate burnup steps that are used to compute the k values and the average cross sections with updated neutron spectra. Table 2 shows the results for a three step burnup calculation, up to 60 GWD/TON, for the case of the assembly of Figure 1 with a Pu load equivalent to "1 gram Pu/pebble.

CONCLUSION

The SCALE system has been used to analyze, at the full assembly level, the use of VHTR for the dual purpose of producing energy and burning Pu. The double heterogeneity characteristics of this kind of reactor makes necessary the calculation of the Dancoff factors associated with the small fuel kernels. This was done with the code MCNP for a large variety of arrangements, pebble beds and prismatic fuel with particular care on defining the situation under which the Dancoff factor calculations are independent of the boundary conditions. SCALE calculations were then performed by mimicking the SAS2 burnup sequence with the extra step imposed by the assembly level geometry. Comparisons with MCNP calculations are shown at BOL; they show reasonable

agreements which give confidence in the procedures used to compute the Dancoff factors

REFERENCES

1. SCALE 5: Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers. Available from the RSICC as package CCC-725.
2. J. F. BRIESMEISTER, Ed., "MCNP™ – A General Monte Carlo N-Particle Transport Code, Version 4C," LA-13709-M (2000).

Table 1
Dancoff Factors for a Central Fuel Kernel Located in a Central Compact in a Prismatic Assembly

Case	Radius Kernel Microns	Density Kernels/cc	Dancoff Factor	
			Simple Cubic Lattice	Body Centered Cubic Lattice
1	100	413.357	0.1560	0.1599
2	150	122.472	0.1099	0.1045
3	200	51.670	0.0939	0.0793
4	250	26.455	0.0555	0.0597

Table 2
Burnup Dependent Calculations with the SCALE System for VHTR Assembly with Pu Load Equivalent to "1 gram Pu/pebble", 100 micron kernel DF=0.16

k=k (Burnup)

Calculation	0	20 GWD/TON	40 GWD/TON	60 GWD/TON
Fuel Kernel	1.145	1.116	1.097	1.080
Compact Cell	1.297	1.275	1.257	1.241
Assembly	1.357	1.335	1.318	1.302