DEPLETED REACTOR ANALYSIS WITH MCNP-4B

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Introduction

Monte Carlo neutronics calculations are mostly done for fresh reactor cores. There is today an ongoing activity in the development of Monte Carlo plus burnup code systems made possible by the fast gains in computer processor speeds.

In this work we investigate the use of MCNP-4B [1] for the calculation of a depleted core of the Soreq reactor (IRR-1). The number densities as function of burnup were taken from the WIMS-D/4 [2] cell code calculations. This particular code coupling has been implemented before ([3]-[5]).

The Monte Carlo code MCNP-4B calculates the coupled transport of neutrons and photons for complicated geometries.

We have done neutronics calculations of the IRR-1 core with the WIMS and CITATION codes in the past [6]. Also, we have developed an MCNP model of the IRR-1 standard fuel for a criticality safety calculation of a spent fuel storage pool [7].

MCNP representation of the Soreq reactor core

We built a MCNP model of the whole reactor for one of the IRR-1 core configurations with the control rods at their logged axial positions. We did extensive use of the MCNP *universe* concept (geometries within geometries) and of the construction of depleted fuel assembly templates, in order to simplify the input generation and prevent errors. We obtained the depleted fuel number densities from WIMS-D/4 burnup calculations.

We wrote a utility code, W2MCNP, in order to perform the following tasks: (1) read the burnup dependent fuel number densities generated by WIMS; (2) interpolate the number density table for each assembly at its nominal burnup; (3) convert to the MCNP material labels; (4) update the {iodine-135, xenon-135} and {promethium-149, samarium-149} number densities to account for a decay and buildup period after shutdown; (5) generate MCNP material cards for each assembly at its nominal burnup.

Figure 1 shows the U-235 loading map of core 950701, which we analyze in the present work. Flux measurements at 100 kW were done on July 4, 1995, after a 2 day cooling period.

Additional rows of graphite elements and water are omitted from the picture. On all four sides, 20 cm of water are added to correspond to the core being located in the large reactor pool. The end boxes on the top and bottom of the assemblies is represented by a homogenized mixture of 25 volume % aluminum and 75 volume % water extending 15 cm beyond the fuel plates; also, 20 cm of water were added on the top [8]. The dummy assembly D-27 has a cadmium cylinder shell for fast neutron irradiation of samples.

FS-14 201.21	FS-23 222.70	**RR** 0.	FS-19 210.61	FS-7 150.19
FS-13 167.86	**FC-5** 98.48	FS-15 168.75	**FC-4** 97.62	FS-16 178.38
FS-9 120.37	FS-28 274.77	FS-45 237.44	FS-18 173.27	FS-22 226.78
FS-6 145.04	**FC-6** 125.54	<u>W</u> 0.	**FC-1** 67.07	FS-25 242.20
FS-17 205.55	FS-26 261.43	FS-11 167.89	FS-27 270.74	FS-12 172.46
D-27 0.	G 0.	FS-30 277.66	G 0.	G 0.

FS	French Standard assembly (20 assemblies)
FC-5	French Control assembly 5 (absorber 82% out)
FC-4	French Control assembly 4 (absorber 82% out)
FC-6	French Control assembly 6 (absorber 82% out)
FC-1	French Control assembly 1 (absorber 81% out)
RR	Regulating rod (52% out)
G	Graphite assembly
<u>W</u>	Water in flux trap location
D-27	Dummy assembly with cadmium
Number	U-235 loading per assembly (grams)

Figure 1: Loading map of array No. 950701

Criticality calculations

A criticality calculation was done for core 950701 with the nominal burnups for all assemblies; and control and regulating rods at the recorded positions. The result obtained with 2,000,000 active neutron histories is:

$$k_{eff} = 1.0225 \pm 0.0005$$

The main reasons for the departure from criticality are:

1. <u>The uncertainty in the nominal assembly burnups</u>. The assembly burnup is being investigated separately [9].

2. <u>Fission product representation in WIMS</u>. We use the 69-group ‘1986’ WIMS Library: it includes burnup data for 35 fission product isotopes and the remainder represented by a pseudo fission product (nuclide 4902). Since there is no such an artificial isotope in the MCNP cross section library, we neglected it; the effect on the criticality, at the WIMS level, is about +700 pcm at 80% burnup.

Recently, the Nuclear Data Section of the International Atomic Energy Agency has released the new WIMSD-IAEA-69 group cross section library [10]. In the documentation [11], the author identifies 131 fission product isotopes. Of these, 45 are treated explicitly and 86 are lumped into a new pseudo fission product. It is expected that the use of this new library will diminish the contribution of the pseudo fission product (as well as provide an overall improvement in the cross section data base).

The calculation of the total flux tallies at the central flux trap is in progress. The tallies will be compared to the experimental data obtained by gold activation.

Conclusions

We developed an MCNP model of the IRR-1 reactor.

We did a core criticality calculation for configuration 950701, using fuel number densities from WIMS cell burnup calculations.

In the future, the use of the WIMSD-IAEA-69 library (in conjunction with the updated code WIMSD-5B) should diminish the contribution of the pseudo fission product, as well as provide an improved cross section set.

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