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The DIMPLE S01A critical assembly

Results of MCNP4A calculations

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Abstract

The DIMPLE S01A critical assembly at AEA Winfrith (a light-water moderated research reactor) was modelled in MCNP4A. Calculations were performed with the Monte Carlo code MCNP4A, using cross-section data from the EJ2-MCNPlib library. The work was done in the framework of validation of this library (which is based upon the JEF-2.2 evaluation) for light-water reactor applications.

Using the experimental critical moderator level the values of k_{eff} and three central reaction rate ratios ($F9/F5$, $C8/F5$ and $F8/F5$) were calculated.

Good agreement between measured and calculated values of k_{eff}, $F9/F5$ and $C8/F5$ is obtained. The value of $F8/F5$ is underpredicted by 10%. Similar results were obtained using ^{238}U cross-section data from the ENDF/B-VI.2 evaluation. Therefore, this latter deviation might be due to experimental (normalisation) errors in the determination of $F8$.

Keywords

Criticality calculations
Benchmark calculations
LWR
MCNP4A
EJ2-MCNPlib
JEF-2.2 cross-section data

CONTENTS

1. INTRODUCTION

Monte Carlo analyses of the neutronic behaviour of an LWR in combination with continuous-energy cross-section data are an attractive tool to give a detailed description of a static LWR core. Very few limitations exist in the field of the geometric modelling of a problem. Furthermore, almost every detail of the original nuclear data evaluation is retained in the cross-section library and self-shielding in the resolved resonance range is explicitly taken into account. However, validation of nuclear data with relevant experimental data is required.

A good benchmark for LWR criticality and reactivity analyses is the DIMPLE S01A benchmark [1], in which a standard LWR core is modelled.

In this report the results are given of benchmark calculations performed with the Monte Carlo neutron transport code MCNP4A [2] for the DIMPLE S01A geometry using cross-section data from the EJ2-MCNPlib library, which is based on the European JEF-2.2 evaluation.

A description of the geometry of the DIMPLE S01A assembly is given in chapter 2. In chapter 3 details on the calculations are presented. The detailed model, with which the DIMPLE S01A assembly is described, is illustrated in chapter 4. Results of the calculations are given in chapter 5. Finally, in chapter 6 conclusions from this work are drawn.

2. GEOMETRY

The DIMPLE reactor, located at the Winfrith site of AEA Technology, is a versatile, light-water moderated zero power research reactor used to investigate performance, safety and safeguards issues relevant to the entire nuclear fuel cycle.

The S01A assembly comprises 1565 3% enriched uranium dioxide fuel pins arranged on a square pitch of 1.32 cm to provide a cylindrical, light-water moderated core 59 cm in diameter and just under 50 cm high. It is a high-leakage assembly, with over 20% of the neutrons leaking from the core.

A detailed description of the DIMPLE S01A geometry and of the composition of the assembly is given in [1].

3. METHOD

3.1 Calculational procedure

Neutron transport calculations for this benchmark were performed using the Monte Carlo code MCNP4A [2]. Continuous-energy cross section data were used, which allow for a very detailed simulation of the neutron transport.

The calculation of k_{eff} and the reaction rates was divided into three subsequent steps:

1. As an initial source S_0 an isotropic point source in the central fuel pin was taken. Using a batch size of 400 neutrons, in 400 cycles a geometrically converged source S_1 was produced, which was used in the second step of the calculation.
2. Starting with S_1 a second calculation was performed, using a batch size of 4000 neutrons. In 200 cycles the initial source S_2 for the final step was produced.
3. In the final run the batch size was extended from 4000 to 40000 neutrons, which corresponds to an average of 25 neutrons for each fuel pin in the S01A assembly for each cycle. The values of k_{eff} and the reaction rates were calculated in 320 cycles.

3.2 Cross sections

JEF-2.2 based cross-section data for all isotopes were taken from the EJ2-MCNPlib library [3], processed at ECN Petten.

4. CALCULATIONS

4.1 Geometrical model

The model for the DIMPLE S01A assembly used in the MCNP4A calculations comprises four important components of the core (see fig. 4.1):

- the fuel pins,
- the upper lattice plate,
- the lower lattice plate and
- the part of the fuel support plates and fuel beam bases that sustains the lower lattice plate.

All dimensions and material data were taken from [1].

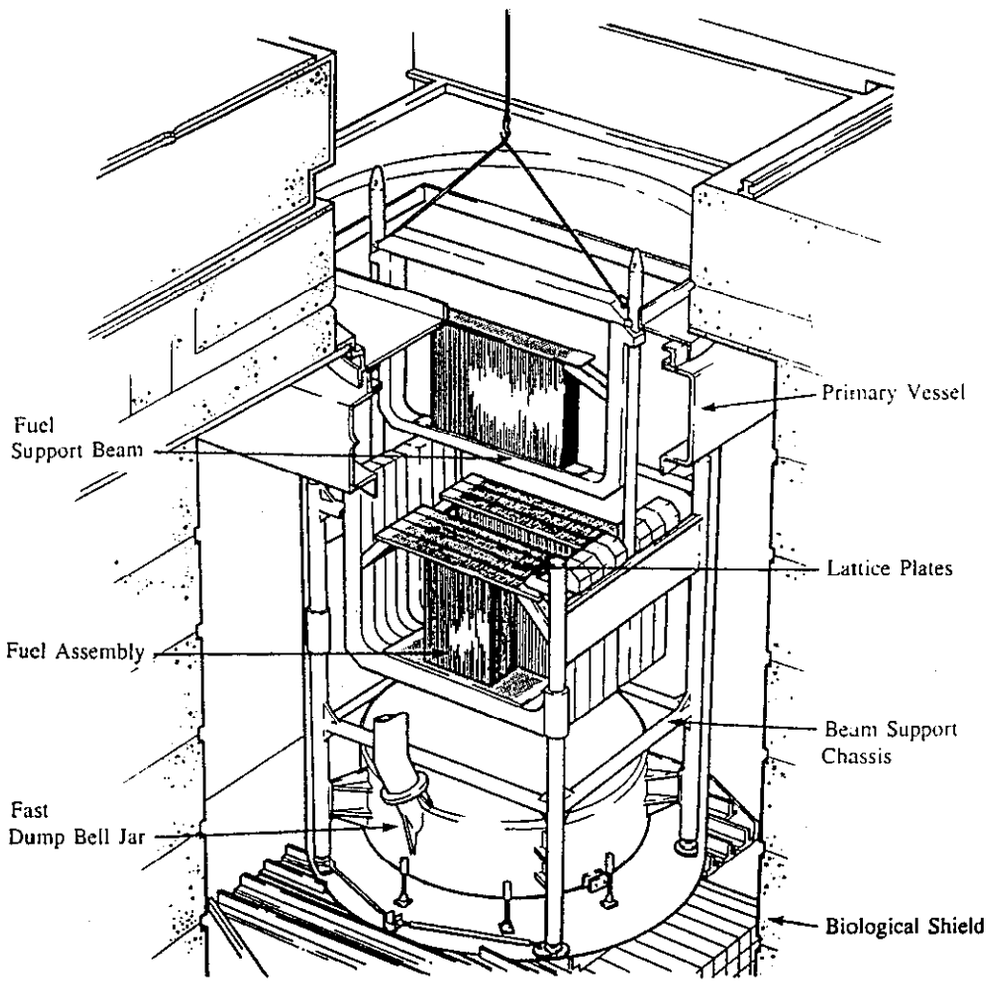

Figure 4.1 Schematic lay-out of S01A assembly within DIMPLE primary vessel (figure taken from [1]).

For the S01A assembly, the critical water level is 49.26 cm at room temperature. This distance is measured with respect to the bottom of the fuel in the fuel pins.

The geometrical model is illustrated in figs. 4.2 to 4.8. All figures relate to the most extended geometrical model, as mentioned below. Dimensions in the figures are in cm. In fig. 4.2 and 4.3 horizontal cross sections of the geometry are given. Vertical cross sections of the core geometry are given in figs. 4.4 and 4.5.

In fig. 4.6 a detailed view is presented of the fuel pins, showing four interstitial drainage holes and a fuel pin with fuel, fuel wrapper (aluminium foil) and (stainless steel) cladding.

Details of the upper and lower end plugs (and stainless steel dowels) are given in figs. 4.7 and 4.8, respectively.

In the MCNP-model not the complete reactor was modelled, as only the core region needs to be taken into account for the calculation of k_{eff} and reaction rates. In [4] it is stated, that in the DIMPLE S01A assembly 13 cm of water forms an effectively infinite reflector. Thus, features more than 13 cm away from the edge of the outermost pin could be ignored. The mean free path for fission neutrons in water is approx. 3.5 cm, hence the value of 13 cm might be too small. In order to verify the assumption, 3 different geometrical models were constructed:

1. the standard model, in which the core is surrounded by at least 13 cm of water;
2. the extended model, in which the amount of water in the radial direction is increased, so that the core is surrounded by at least 28 cm of water;
3. the most extended model, which equals the extended model except for the fact that a layer of water with a thickness of 13 cm is added below the fuel support plates.

MCNP-calculations were performed for each of these 3 geometries. The results of the calculations are given in chapter 5.

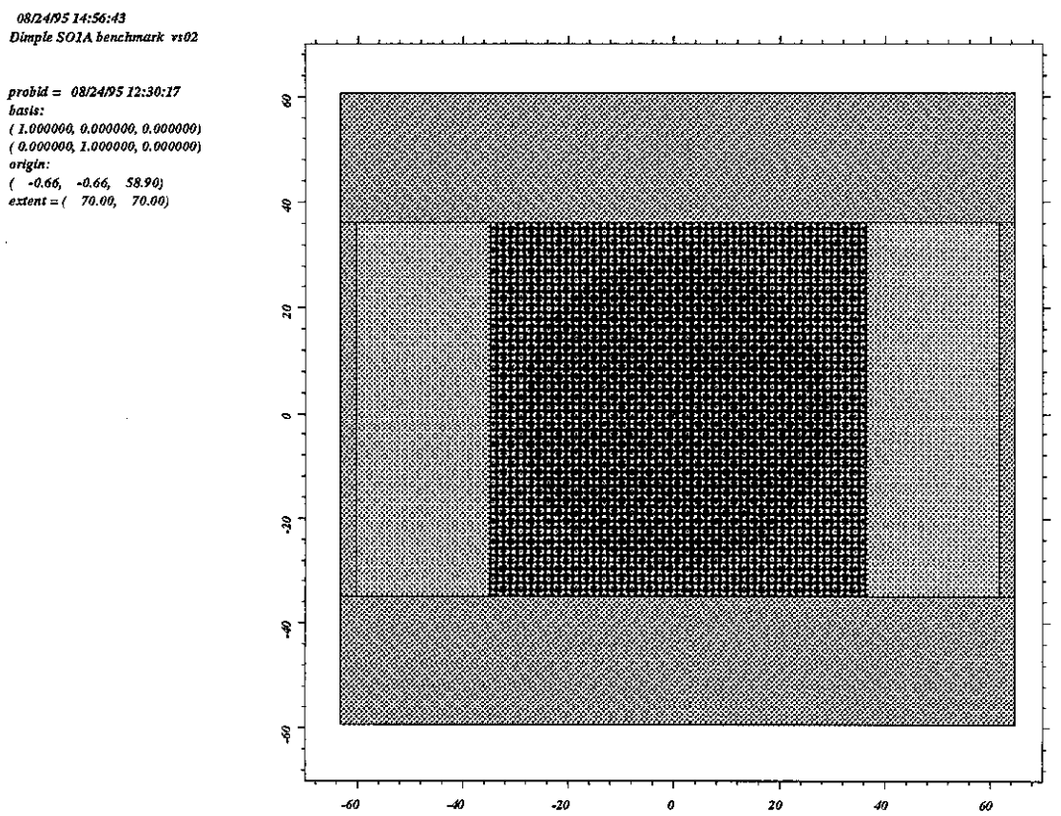

Figure 4.2 MCNP4A model, horizontal cross section of upper lattice plate. Overview of S01A assembly.

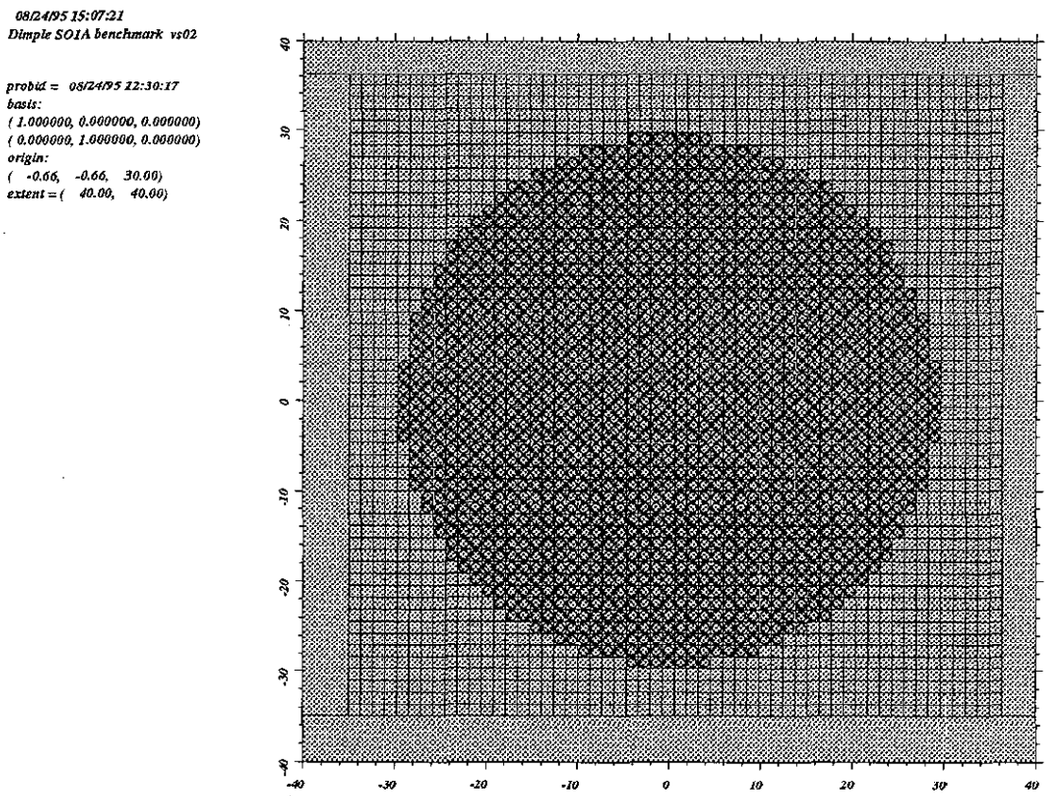

Figure 4.3 MCNP4A model, overview of S01A assembly. Horizontal cross section at 30 cm above fuel base.

4.2 Reaction rate calculations

Reaction rates for the central core position were measured by foil activation techniques. Foils were inserted between the fuel pellets of fuel pins inside the central area of the core lattice [5]. The central core measurements were performed are at mid water height (measured from the base of the fuel [6]).

In the MCNP4A calculations, reaction rates were determined in the central fuel pin and 20 surrounding fuel pins. These positions correspond to the positions of the activation foils during the experiments. In order to study the axial behaviour of the reaction rates, reaction-rate calculations were performed in 6 axial bins. The bins are separated by the horizontal lines in the core region in fig. 4.5.

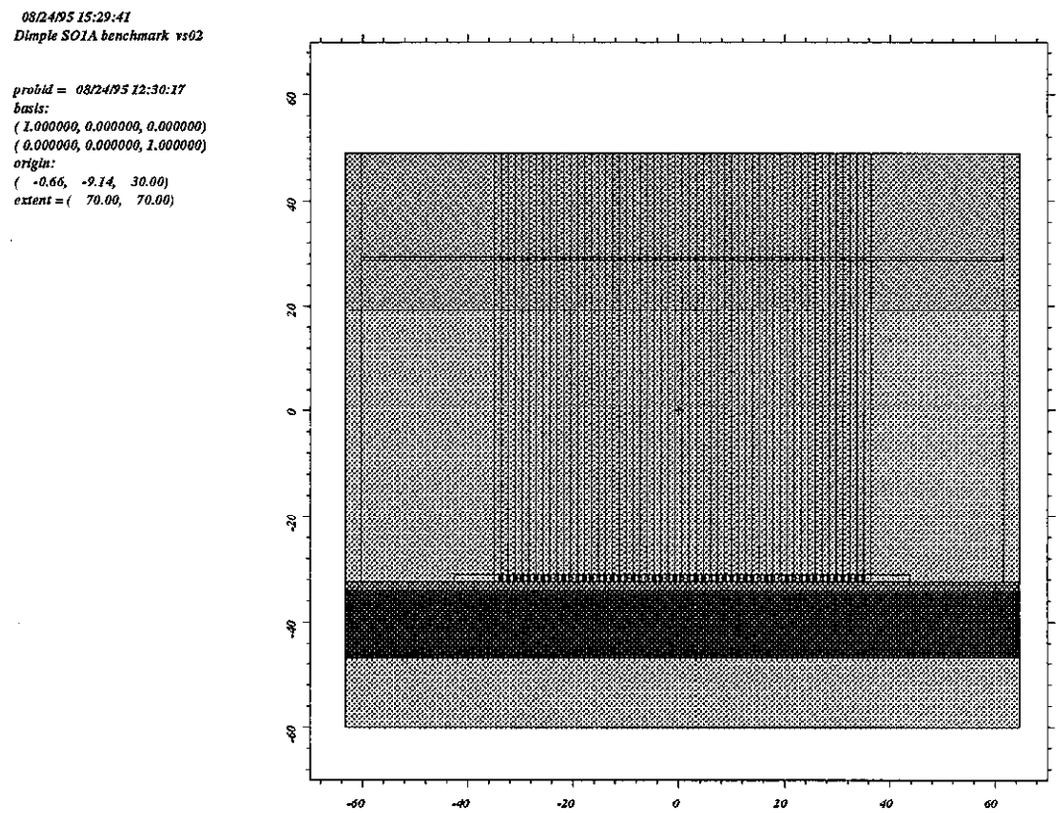

Figure 4.4 MCNP4A model, overview of S01A assembly. Vertical cross section along fuel support plate.

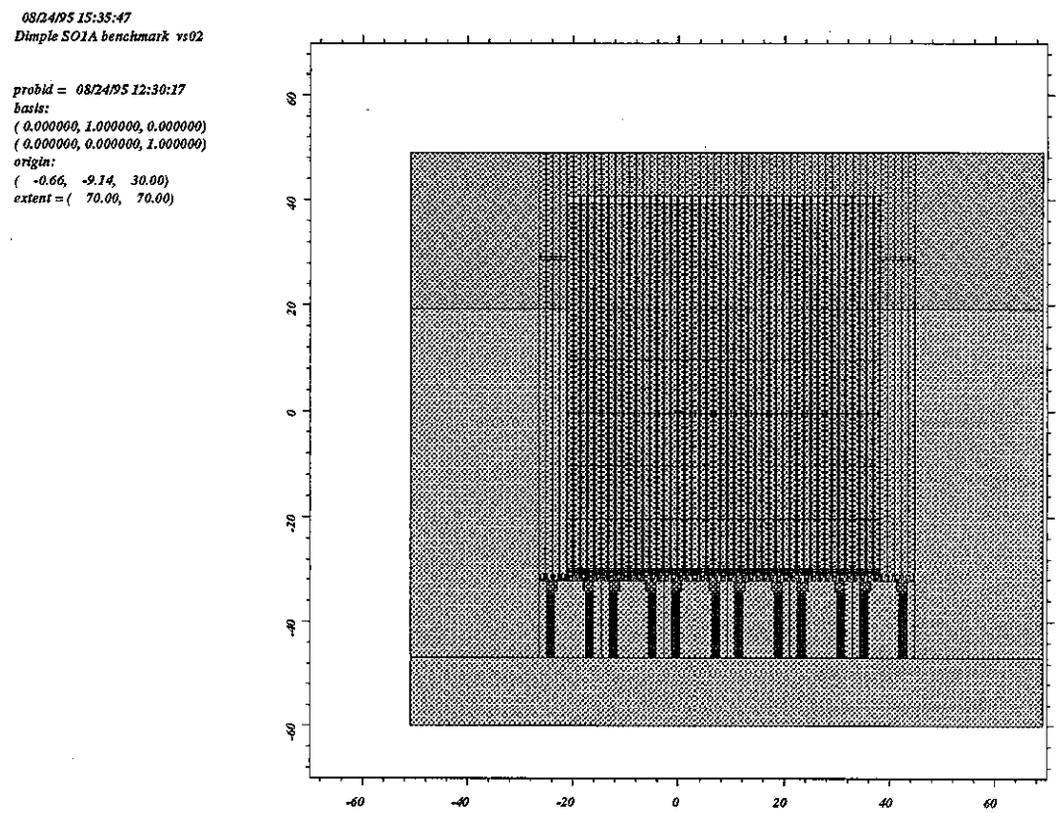

Figure 4.5 MCNP4A model, overview of S01A assembly. Vertical cross section across fuel support plates and fuel beam bases.

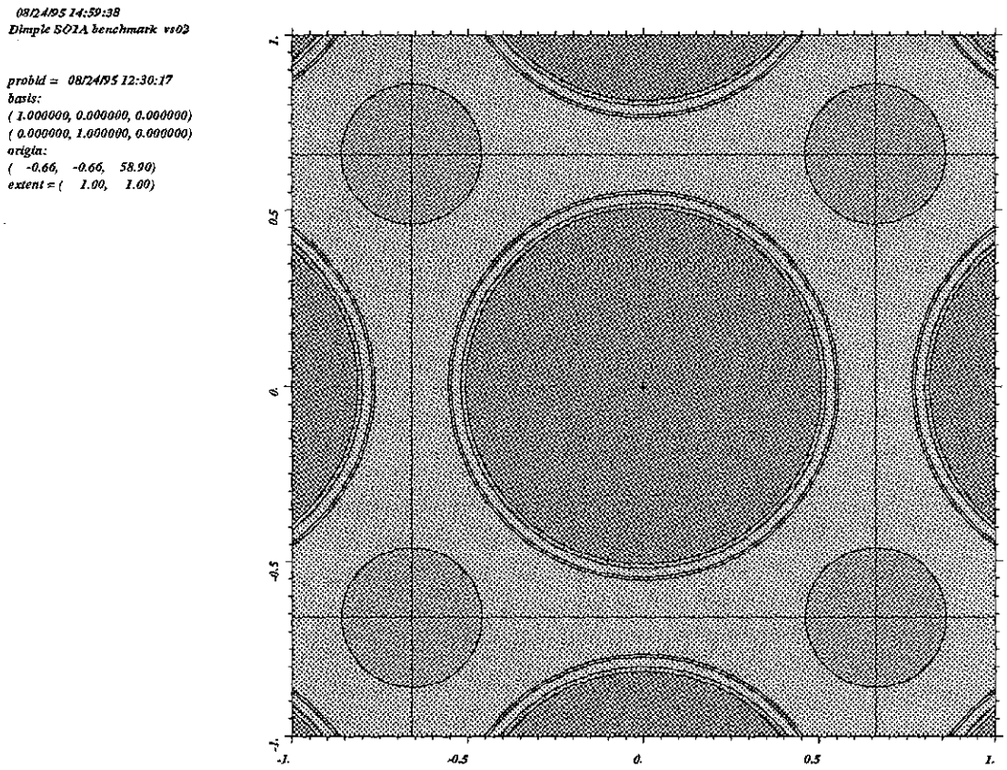

Figure 4.6 MCNP4A model, horizontal cross section through upper lattice plate. View of fuel pins and upper lattice plate drainage holes.

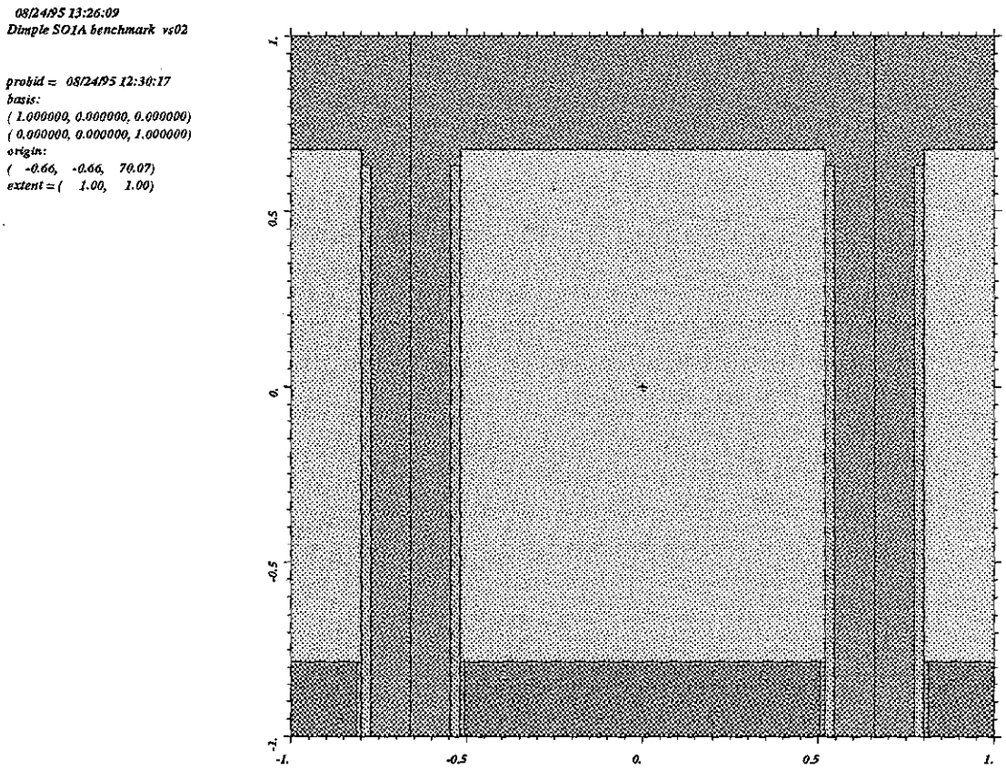

Figure 4.7 MCNP4A model, vertical cross section of fuel pin and upper end plug.

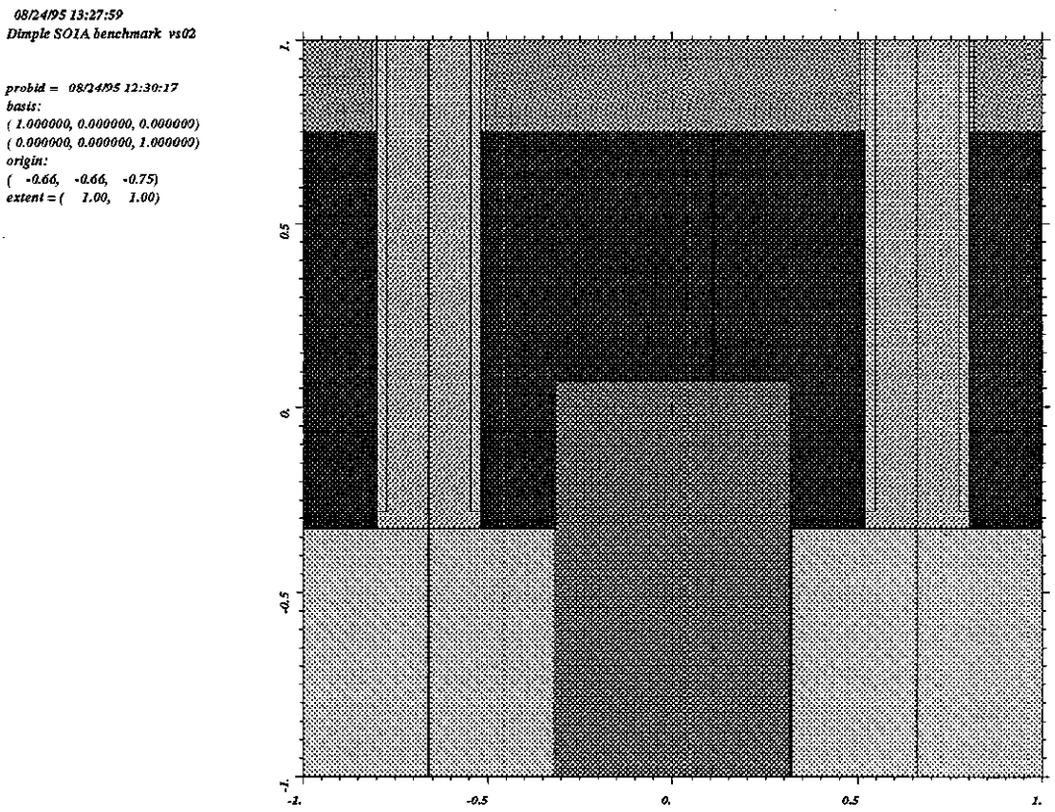

Figure 4.8 MCNP4A model, vertical cross section of fuel pin and lower end plug, pin dowel and lower lattice plate.

5. RESULTS AND DISCUSSION

Calculated values of k_{eff} for the three geometries given in section 4.1 are given in table 5.1. From these values the assumption from [4] is verified: there is no influence on the value of k_{eff} due to the addition of extra water around the core. In all models an excellent agreement between measured and calculated value of k_{eff} is observed.

In table 5.2 reaction rate ratios are given as calculated in the most extended geometrical model. The ratios were calculated in axial bins in order to study the axial dependence. For each level, the reaction rates are averages of the 21 central fuel pins. Reaction rate ratios were measured in the core centre, which corresponds to axial bin 2.

Measured and the calculated (in axial bin 2) values of the reaction rate ratios are compared in table 5.3. From this table it is clear that a good agreement is obtained between measured and calculated values of $F9/F5$ and $C8/F5$. However, the measured and calculated values of $F8/F5$ disagree, as the experimental value is underpredicted by 10%. Although the experimental uncertainty in the determination of $F8/F5$ is rather large, the difference is clearly outside the error-band.

This difference may be due to

- errors in cross-section data;
- errors in the calculational model;
- experimental errors in the determination of $F8/F5$

Because of the good agreement between measured and calculated values of $F9/F5$ and $C8/F5$ errors in the cross-section data for ^{235}U may be excluded. Hence, if the difference is due to errors in cross-section data it should be due to errors in cross-section data for the fission cross section of ^{238}U. Using ENDF/B-VI.2 cross-section data for ^{238}U instead of JEF-2.2 cross section data leads to statistically indistinguishable results for the reaction rate ratios. This implies that errors in cross-section data are an unlikely source for the observed difference. One should bear in mind, however, that cross sections in the resonance ranges in the JEF-2.2 and ENDF/B-VI.2 ^{238}U evaluations are essentially identical up to $E_n = 149$ keV, which decreases the sensitivity to ^{238}U fission cross section in a thermal spectrum.

Errors in the calculational model may safely be excluded because of the good agreement obtained between the measured and calculated values of k_{eff}, $F9/F5$ and $C8/F5$.

Therefore, the remaining possibilities for the discrepancy observed are either experimental errors in the determination of $F8/F5$ or errors in cross-section data of ^{238}U. Further analysis is needed in order to make a firm decision.

Table 5.1 *Values of k_{eff} as calculated in the three geometries given in section 4.1*

Geometry	$k_{eff} \pm \sigma$
standard	1.0000 ± 0.0002
extended	1.0000 ± 0.0002
most extended	1.0000 ± 0.0002

Table 5.2 *Calculated reaction rate ratios based on reaction rates averaged over the central 21 fuel pins. The positions of the axial bins are indicated in fig. 4.5.*

axial bin	distance to fuel base [cm]	$F8/F5 \pm\sigma$ [%]	$F9/F5 \pm\sigma$ [%]	$C8/F5 \pm\sigma$ [%]
0	0.00 - 9.63	2.57E-03 ± 0.83	2.17 ± 0.96	1.90E-02 ± 0.96
1	9.63 - 19.63	2.73E-03 ± 0.67	2.21 ± 0.79	2.02E-02 ± 0.78
2	19.63 - 29.63	2.73E-03 ± 0.63	2.21 ± 0.73	2.03E-02 ± 0.72
3	29.63 - 39.63	2.73E-03 ± 0.66	2.22 ± 0.78	2.01E-02 ± 0.77
4	39.63 - 49.26	2.81E-03 ± 0.87	2.21 ± 1.0	2.12E-02 ± 1.0
5	49.26 - 69.285	1.10E-02 ± 1.5	2.28 ± 2.6	4.40E-02 ± 1.6

Table 5.3 *Comparison of measured and calculated central reaction rate ratios. The calculated data are based on reaction rates averaged over the central 21 fuel pins.*

ratio	measured $\pm\sigma$ [%]	calculated $\pm\sigma$ [%]	$C/E \pm \sigma$ [%]
F8/F5	3.02E-03 ± 3.4	2.73E-03 ± 0.63	0.904 ± 3.5
F9/F5	2.19 ± 0.9	2.21 ± 0.73	1.01 ± 1.2
C8/F5	2.03E-02 ± 0.5	2.03E-02 ± 0.72	1.00 ± 0.9

6. CONCLUSIONS

In this report the results are presented of a detailed Monte Carlo analysis of the DIMPLE S01A critical assembly. The analysis was carried out with MCNP4A, whereas JEF-2.2 based cross-section data from the EJ2-MCNPlib library were used.

Good agreement between measured and calculated values is obtained for k_{eff} and the reaction rate ratios $F9/F5$ and $C8/F5$. This validates the combination MCNP4A/EJ2-MCNPlib library for LWR-applications.

A discrepancy is observed between the measured and calculated value of $F8/F5$ (the measured value is underpredicted by 10% in the calculations). This might be due to experimental errors. Further analysis is needed to solve this problem.

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