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PROPOSAL FOR A SECOND STAGE OF THE BENCHMARK ON POWER DISTRIBUTIONS WITHIN ASSEMBLIES.

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1. Physical motivations

A first stage of this benchmark was proposed in 1991 and was essentially devoted to "whole core calculations" in the sense that one started with neutronic parameters already smeared ("homogenized") for each pin cell type and collapsed into a fast and a thermal broad group.

In fact the real problem is much more complex because the method for deriving such parameters is not obvious.

Generally, one starts by calculating the assemblies using transport codes (for example by solving the Integral transport equation) with an heterogeneous description of the pin cells and a fine treatment of the energy variable.

The way to obtain 2 group cell parameters from the calculation for an assembly (or even a group of assemblies) is not trivial, because :

a) The spectrum is very space-dependent, especially near the boundary between MOX and UO_2 assemblies ;

b) the definition of cell by cell diffusion coefficients (radial and axial) is not always well founded ;

c) The procedure used to derive homogeneous "equivalent" cell parameters can be a source of error.

Furthermore, a calculational scheme has to be considered as a whole.

For example, the ways to derive cell of assemblies parameters are generally not independent :

- Neither of the adopted type of heterogeneous cell transport solution,
- Nor of the scheme adopted for the whole core calculations.

There might be some "equivalent procedures" (or discontinuity factors) whose role is to modify the cell or assembly parameters in order to compensate for possible errors due to homogenization, mesh effects, etc.

For these reasons, we think that it would be very interesting to compare the results obtained by the different participants starting from the beginning, that is to say from the detailed geometrical and physical data.

Of course, one of the main drawbacks in doing this could be the difficulty to separate the discrepancies due to the differences in the cross-sections (which is not the purpose of this benchmark) from those in the models.

However, we think that, except for unexpected errors in the libraries, the impact of differences in the cross-sections on the calculated pin-wise power distributions should be quite small compared to other sources of error in the modelization of the problem. Hence, our proposal is as follows.

2. Geometrical and physical data

The detailed geometries of the unit pin cells involved in the problem are given in Table 1

- The geometries of the two assemblies are given in Figures 1 and 2.
- The core geometry is given in Figure 3.
- The isotopic composition for each medium is defined in Table 2.
- The temperature is constant and equal to 20°C.
- Vacuum boundary condition is assumed at the external limit of the reflector.
- For 2D calculations, it is assumed that the extrapolated height of the core is 95 cm (80 + 2x7.5 cm), which leads to an axial buckling equal to 1.094 10^{-3} cm^{-2}
- In the central cell of each assembly, the response of a miniature fission chamber is simulated by the fission rate of one atom of U-235 in the water inside the central guide-tube.

3. Calculations and results

In order to check the effect of significant discrepancies due to differences in nuclear data, we ask the participants to perform four unit cell calculations and to provide the following answers :

3.1 Cell Calculations

For each of the fuel cells (MOX4.3 - MOX7.0 - MOX8.7 and UO_2), a critical buckling will be determined and the participants will give the following results :

- ➢ B^2
- ➢ K_∞ (Ratio of production rate to absorption rate in the spectrum taking into account the leakage)
- ➢ M^2 defined as $(1=K_\infty/(1+M^2B^2)$
- ➢ Reaction rates per isotope (1 group and 3 groups involving the 5.53 keV and 4 eV boundaries).

If it is possible and in order to be able to compare with MONTE-CARLO solutions, cylindrical and square cell geometries can be tested.

3.2. *Core Calculations.*

The participants will give the following results :

- K-eff
- Pin by pin group integrated fission rates (and not power) on 1/8 of the geometry, including U-235 fission rate in the central cells of each assembly.
- Integral fission rate per assembly.

4. GENERAL INFORMATION

The participants must provide a short description of their calculational scheme.

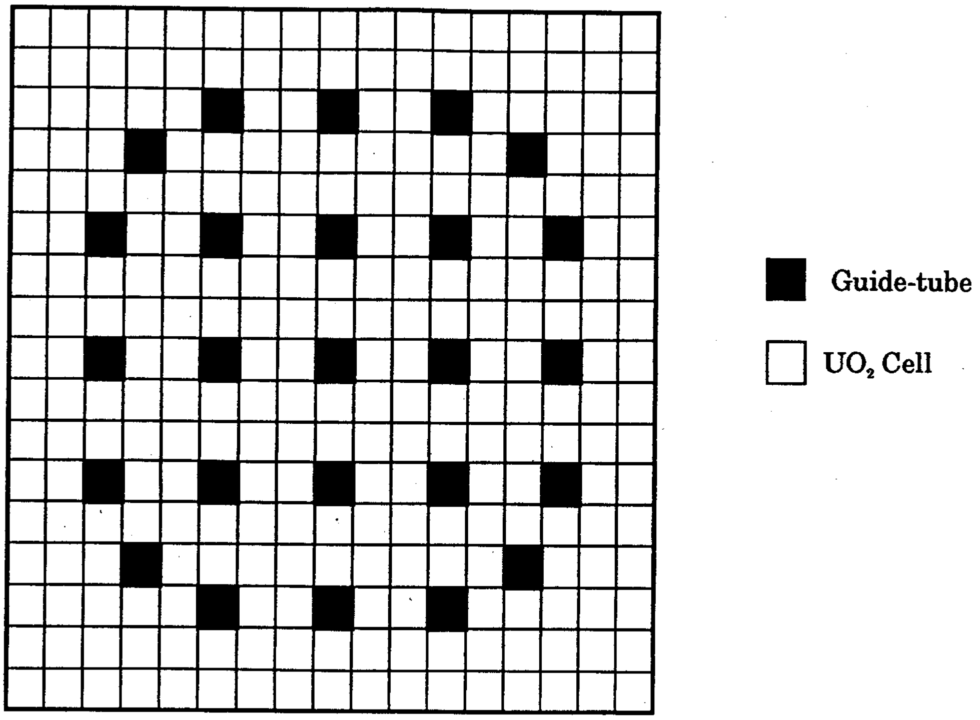


Figure 1 ***UO_2 Assembly***

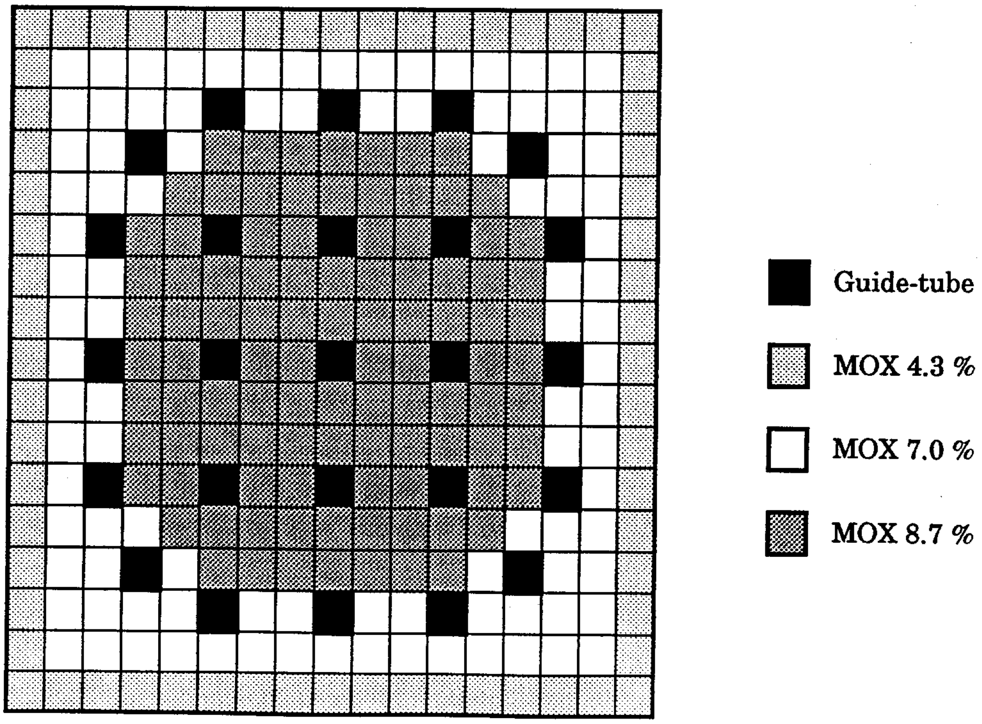


Figure 2 ***MOX Assembly***

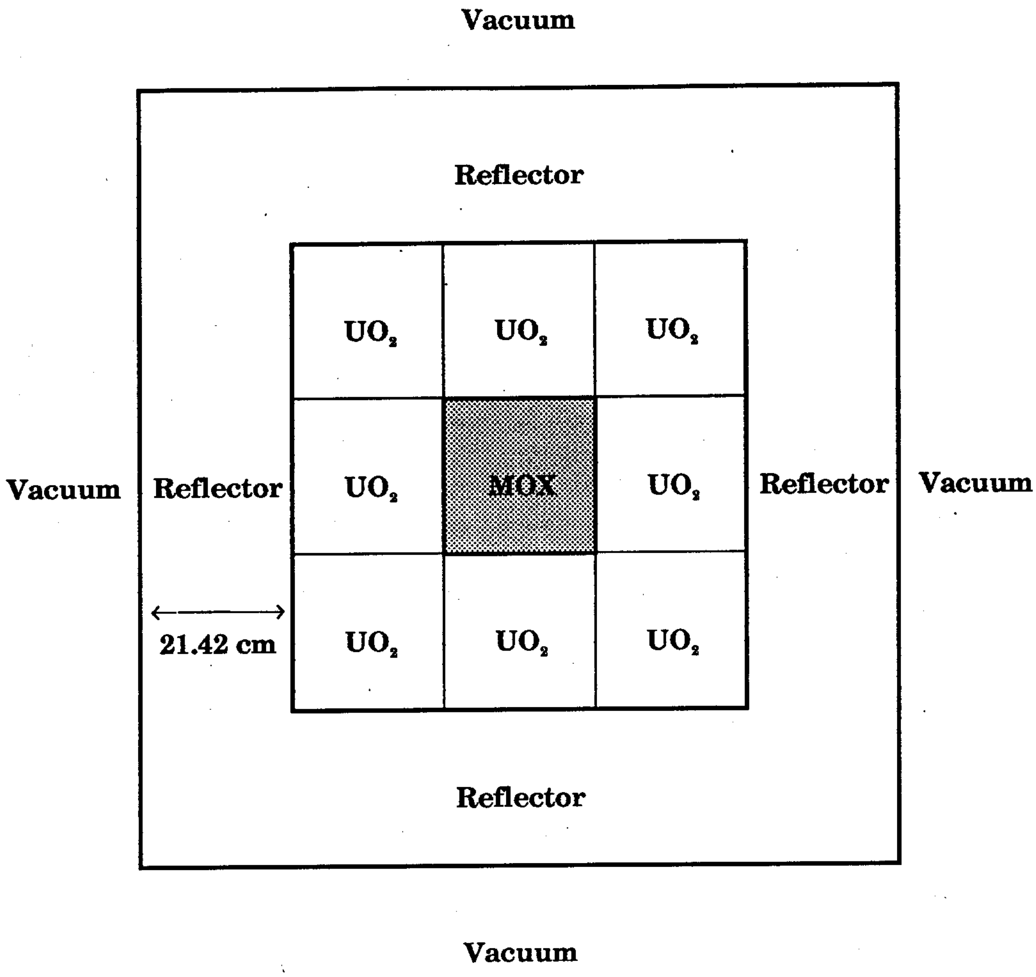


Figure 3 : Reflected Core Geometry

*Table 1 **Cell geometries***

Fuel Cells : MOX 4.3% - MOX 7.0% - MOX 8.7 % and UO_2

Medium	External Radius
Fuel	0.4095 cm
Void	0.4180 cm
Zirconium Clad	0.4750 cm
Void	0.4850 cm
Aluminium Clad*	0.5400 cm
Moderator	Square lattice pitch = 1.26 cm

* This clad is used to simulate hot conditions at room temperature (decrease of the moderation ratio)

Guide-Tube Cells

Medium	External Radius
Moderator	0.3400 cm
Aluminium Clad	0.5400 cm
Moderator	Square lattice pitch = 1.26 cm

Central guide tube contains: moderator (as defined in Table 2) and 1.0E-8 at/(b cm) of U^{235}.

*Table 2 **Isotopic Distributions for each medium***

Nuclide	Concentrations (10^{24} at/cm^3)						
	MOX 4.3 %	MOX 7.0 %	MOX 8.7 %	UO_2	Moderator	Zr Clad	Al clad
U-235	5.00E-5	5.00E-5	5.00E-5	8.65E-4			
U-238	2.21E-2	2.21E-2	2.21E-2	2.225E-2			
Pu-238	1.50E-5	2.40E-5	3.00E-5				
Pu-239	5.80E-4	9.30E-4	1.16E-3				
Pu-240	2.40E-4	3.90E-4	4.90E-4				
Pu-241	9.80E-5	1.52E-4	1.90E-4				
Pu-242	5.40E-5	8.40E-5	1.05E-4				
Am-241	1.30E-5	2.00E-5	2.50E-5				
O	4.63E-2	4.63E-2	4.63E-2	4.622E-2			
H_2O					3.35E-2		
Nat. B					2.78E-5		
Nat. Zr						4.30E-2	
Al-27							6.00E-2

Annex 1

Results to be reported in Phase II of the Benchmark on Power Distributions within Assemblies

1) Cell calculations

The following results should be reported in the format as given in tables below:

cell type	B^2	k∞	M^2
MOX 4.3%			
MOX 7.0%			
MOX 8.7%			
UO_2			

Table 1: Cell calculations - key parameters

Reaction rates per isotope (absorptions + fission rates + production rates $\nu\Sigma_f\phi$) to be given in one group and in three groups (E_1 <4 eV, 4eV < E_2 < 5 keV, E_3 > 5 keV) in the following tables.

	MOX 4.3%											
	group fission rate				group absorption rate				group production rate			
Nuclide	g=1	g=2	g=3	Total	g=1	g=2	g=3	Total	g=1	g=2	g=3	Total
U^{235}												
U^{238}												
Pu^{238}												
Pu^{239}												
Pu^{240}												
Pu^{241}												
Pu^{242}												
Am^{241}												
$O^{(fuel)}$												
H_2O(Moder)												
Boron(Moder)												

Table 2.1 Cell calculations - absorption and fission rates in three and one energy groups for MOX 4.3% fuel

Table 2.2 → As Table 2.1 but for MOX 7.0 %

Table 2.3 → As Table 2.1 but for MOX 8.7 %

Table 2.4 → As Table 2.1 but for UO_2 fuel

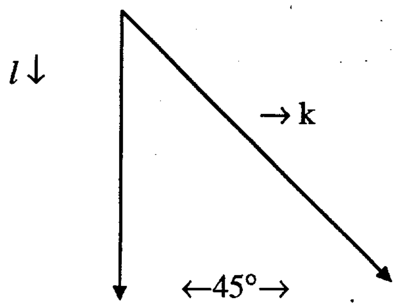
2) Core calculations

The following results are to be reported:

1. k_{eff}

2. Pin-by-pin group integrated fission rates on 1/8 core geometry.

Pins in one quarter of the core form a 27 x 27 matrix. Pins in one-eighth of the core are represented by the lower diagonal of this matrix if the following coordinate system is imposed:



l=1, ..., 27
k=1, ..., 1

Thus, the fission rates F_g(k,l) **summed over all isotopes** in energy group "g" for pins at positions (k,l) are to be reported row-by-row where:

l=1, ..., 27
k=l, ..., l
g=1, 2, 3 ($E_{g=1}$, <4eV, 4eV <$E_{g=2}$ <5keV, $E_{g=3}$ >5kev).

Example:

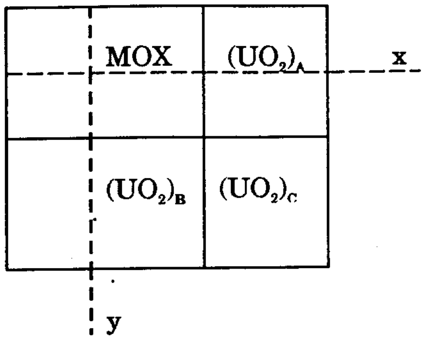
In group g=2, in row l=3, k takes values k=1, k=2, k=3 and hence 3 numbers are to be reported:

$$F_{g=2}(1,3),\quad F_{g=2}(2,3),\quad F_{g=2}(3,3),$$

and so on row-by-row until l=27.

3) Integral fission-rate per assembly

Here one number corresponding to total fission rate in all energy groups in an assembly in 1/4 of the core is to be given according to the graph:



and in the following table:

	Fission Rate Per Assembly			
Nuclide	MOX	$(UO_2)_A$	$(UO_2)_B$	$(UO_2)_C$
U^{235}				
U^{238}				
Pu^{239}				
Pu^{241}				
Am^{241}				

Table 3: Core calculations. Integral fission rates per assembly.

Note: If possible, 2-D square, cylindrical and Monte Carlo solutions should be tested.

Details to be provided about the calculational scheme used

1. Name of participant:

2. Establishment:

3. Name of Code System(s) used:

4. Bibliographic References:

5. Origin of Cross Section Data (e.g. ENDF/B-VI, JEF-2.2, JENDL-3.2 ...):
(describe deviations of standard libraries, e.g. mix from different libraries, details..)

6. Spectral calculations and data reduction methods used:
please describe your scheme, through a graph and explanatory words
provide details about assumptions made

a. resonance shielding: specify method(s) and specify energy range, and the nuclides (actinides, clad, fission products, oxygen, ... unresolved resonance treatment

b. mutual shielding (overlapping of resonances)

c. fission spectra: specify whether only a single spectrum was used or a weighted mix from all fissile nuclides, explain procedure

d. how was the (n,2n) reaction treated?

e. weighting spectrum for scattering matrices, e.g. correction of the out-scatter and self-scatter terms considering the differences between the original weighting spectrum and realistic cell spectrum

7. Number of energy groups used in the different phases:

8. Cell calculation:

a. type of calculation: (i.e. heterogeneous, homogeneous)

b. theory used: (diffusion, transport)

c. method used; (finite difference, finite elements, nodal, Sn(order), collision probability, Monte Carlo, J+/-, other...)

d. calculation characteristics: (meshes, elements/assembly, meshes/pin, number of histories, multi-group, continuous energy ...)

e. reconstruction method:

9. Other assumptions and characteristics:

10. Comments useful for interpreting correctly the results: