CALCULATIONAL BENCHMARK FOR THE ANALYSIS OF SMALL-SAMPLE REACTIVITY EXPERIMENTS

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1. Overview of the benchmark problem

Small-sample reactivity experiments, which are sometimes labeled “pile-oscillation experiments”, refer to the measurement of the effect of a physically small sample of material on the reactivity of a nuclear reactor. These experiments are useful for deriving information about the mechanism of interaction of neutrons with various materials introduced in a nuclear reactor. Such experiments were performed in reactors like ZPR, BFS, DIMPLE and still today in the MINERVE reactor. These experiments are useful to provide characteristics of the reactor himself, and to determine certain properties (e.g. cross sections) of the materials. Concerning this last point, a better knowledge of nuclear data has a major interest in many fields of the nuclear physics like, for instance, the allowing of reactivity credit for spent fuels (also named the “Burn-Up Credit”) which offers both economic and risk incentives.

The experiment generally involves the measurement of small reactivity worth, i.e. considerably less than β, lying in a range from 10^{-8} to 10^{-3} dk/k (0.001 to 100 pcm). Consequently, extremely high precision is required for the interpretation of such experiments by calculation tools. The objective of this benchmark is to verify that simple models can be used to calculate small reactivities, and to enlighten specific problems associated with the interpretation of such experiments. The finality of this work is to take into account the experimental validation provided by programs like CERES (carried out in the MINERVE and DIMPLE reactors between 1993 and 1995) for BUC studies.

The benchmark is separated in four phases which can be treated separately. They are described below:

- Phase I of the benchmark deals with a preliminary analysis on a single PWR fuel cell, perturbated with small amounts of BUC nuclides. 25 isotopes are considered, covering the major fission products and actinides. They represent more than 90% of the reactivity loss of a 4-cycle irradiated assembly fuel. Concentrations have been chosen to correspond to the inventory of a 40 GWd/t spent fuel.
- Phase II of the benchmark is a more complex survey with a natural UO_2 fuel cell of interest surrounded by PWR fuel cells in a reduced two-dimension 7x7 lattice of the core with reflexive conditions on each boundary. This simplified model is supposed to emphasize specific problems in the analysis of small-sample reactivity worth measurements (dancoff effect in space-dependent self-shielding, 2D heterogeneous calculation…). In this phase, concentrations of separated BUC nuclides will be derived from mass amounts used in the CERES program, to provide a reactivity effect of a few tens of pcm in the experiment.
- Phase III of the benchmark is similar to Phase II, with the exception that the axial dimension of the problem is treated. A finite height for the core lattice is imposed with leakage conditions on the lower and upper plan to provide a realistic axial distribution of the neutron flux along the height of the oscillation sample. In this way, questions associated with leakage reactivity effects and “finite-size effect” are taking into account. Reflexive boundaries are maintained in the radial direction.
- Phase IV of the benchmark is similar to Phase III, with the exception that the radial dimension of the problem is treated. A simplified model of the whole core of the reactor is defined as a 49x49 lattice of PWR fuel cells with finite axial dimensions and leakage conditions on axial and radial boundaries. In this way, the real problem can be treated so as to provide a k-effective almost equal to unity.

2. Geometry data

2.1. Benchmark Phase I – single cell calculation

Phase I of the benchmark concerns the analysis of a single fuel cell from a standard 17x17 PWR assembly with an infinite axial dimension. The main geometrical characteristics are described below (see Fig. 1):

PWR cell	
Fuel material	UO_2
Fuel enrichment in ^{235}U	3% w/o
Lattice square pitch	1.26 cm
Fuel radius	0.41 cm
Cladding material	Zircaloy-4
Cladding inner radius	0.418 cm
Cladding outer radius	0.475 cm
Fuel/clad inter-space material	Air
Moderator material	Light water
Material Temperature	300 K
Radial boundary conditions	Reflective

2.2. Benchmark Phase II – 7x7 pattern calculation (axially infinite)

Phase II of the benchmark concerns the analysis of a natural UO_2 fuel cell with an infinite axial dimension, centered in a 7x7 lattice of PWR UO_2 fuel cells (same as Phase I). Reflexive conditions are imposed at boundaries. The main geometrical characteristics are described below (see Fig. 2):

Central cell	
Fuel material	UO_2
Fuel enrichment	0.7% w/o
Lattice square pitch	1.26 cm
Fuel radius	0.41 cm
Cladding material	Zircaloy-4
Cladding inner radius	0.418 cm
Cladding outer radius	0.475 cm
Fuel/clad inter-space material	Air
Moderator material	Water
Material Temperature	300 K
Surrounding cells	PWR cell (same as Phase I)
Lattice	
Number of cells	7x7
Radial boundary conditions	Reflective

2.3. Benchmark Phase III – 7x7 pattern calculation (axially finite)

Phase III of the benchmark concerns the analysis of a natural UO_2 fuel sample centered between aluminum cylinders, in a 7x7 lattice of PWR UO_2 fuel cells (same as Phase I). Leakage conditions are imposed on the lower and upper plans of the 7x7 pattern. The main geometrical characteristics are described below (see Fig. 3):

Central cell	Same as Phase II except:
Fuel and clad length	10 cm
Spacer material	AG-3
Spacer radius	0.475
Spacer length	27 cm

Surrounding cells	Same as Phase II except:
Fuel length	64 cm
Cladding length	64 cm
Lattice	Same as Phase II except:
Axial boundary conditions	Leakage (vacuum)

2.4. Benchmark Phase IV – 49x49 pattern calculation (axially finite)

Phase IV of the benchmark concerns the analysis of a natural UO_2 fuel sample centered between aluminum cylinders, surrounded by a core built with a 49x49 lattice of PWR UO_2 fuel cells (same as Phase I) with leakage conditions on each boundary (vacuum). The main geometrical characteristics are described below (see Fig. 4):

Central cell	Same as Phase III
Surrounding cells	Same as Phase III
Lattice	Same as Phase III except:
Number of cells	49x49
Radial boundary conditions	Leakage (vacuum)
Axial boundary conditions	Leakage (vacuum)

3. Material data

3.1. Benchmark Phase I – single cell calculation

Concentrations associated to the PWR fuel cell materials are given below (temperature = 300K):

PWR cell		
Fuel	Atomic density (atoms/barn.cm)	
(UO_2 – 3%)	U-234	6.160E-06
	U-235	6.900E-04
	U-238	2.200E-02
	O-16	4.540E-02
Fuel-clad space	Atomic density (atoms/barn.cm)	
(Air)	N-14	4.000E-05
	O-16	1.000E-05
Clad	Atomic density (atoms/barn.cm)	
(Zircaloy-4)	Zr	4.247E-02
	Fe	1.486E-04
	Cr	7.598E-05
Moderator	Atomic density (atoms/barn.cm)	
(Light Water)	H-1	6.664E-02
	O-16	3.332E-02

Concentrations of each BUC nuclide to be added to the fuel correspond to a burn-up of 40 GWd/t, for a cooling time of one year (temperature = 300K) :

Separated isotopes to be added to the fuel	Atomic density (atoms/barn.cm)	
Case I.0	Reference case (UO_2-3%) without BUC nuclide	
Case I.1	Mo-95	5.300E-05
Case I.2	Tc-99	5.200E-05
Case I.3	Ru-101	5.100E-05
Case I.4	Rh-103	3.000E-05
Case I.5	Ag-109	4.000E-06

Case I.6	Cs-133	5.600E-05
Case I.7	Nd-143	3.700E-05
Case I.8	Nd-145	3.100E-05
Case I.9	Sm-147	4.900E-06
Case I.10	Sm-149	1.780E-07
Case I.11	Sm-150	1.400E-05
Case I.12	Sm-151	6.700E-07
Case I.13	Sm-152	5.400E-06
Case I.14	Eu-153	4.800E-06
Case I.15	Gd-155	1.000E-07
Case I.16	U-235	1.800E-04 (depleted U5 replace initial content)
Case I.17	U-236	1.000E-04
Case I.18	Np-237	1.500E-05
Case I.19	Pu-238	5.600E-06
Case I.20	Pu-239	1.400E-04
Case I.21	Pu-240	5.500E-05
Case I.22	Pu-241	3.600E-05
Case I.23	Pu-242	1.400E-05
Case I.24	Am-241	2.800E-06
Case I.25	Am-243	3.400E-06
Case I.26	B-10	2.700E-06 (calibration B10)
Case I.27	Sum of all the 25 BUC nuclides (cases I.1 to I.25)	

3.2. Benchmark Phase II – 7x7 pattern cell calculation (axially infinite)

Concentrations associated to the PWR fuel cells of the 7x7 lattice are identical to the ones of Phase I. Concentrations of the central cell materials are given below (temperature = 300K):

PWR cells	Same as Phase I	
Central UO_2 cell		
Fuel	Atomic density (atoms/barn.cm)	
(UO_2 – 0.7%)	U-234	1.575E-06
	U-235	1.764E-04
	U-238	2.375E-02
	O-16	4.785E-02
Fuel-clad space	Same as PWR cells	
Clad	Same as PWR cells	
Moderator	Same as PWR cells	

Concentrations of the BUC nuclides to be added to the central UO_2 fuel are typical values encountered in small sample reactivity experiments, to provide reactivity effects of a few tens of pcm on the whole core (temperature = 300K). Concerning the main FP absorbers, they are about 10 times higher than FP concentrations in LWR spent fuels:

Separated isotopes to be added to the central fuel	Atomic density (atoms/barn.cm)	
Case II.0	Reference case (UO_2-0.7%) without BUC nuclide	
Case II.1	Mo-95	4.300E-03
Case II.2	Tc-99	2.100E-03
Case II.3	Ru-101	6.000E-03
Case II.4	Rh-103	4.300E-04
Case II.5	Ag-109	4.400E-04
Case II.6	Cs-133	1.600E-03
Case II.7	Nd-143	4.400E-04
Case II.8	Nd-145	1.800E-03

Case II.9	Sm-147	7.400E-04
Case II.10	Sm-149	1.900E-06
Case II.11	Sm-150	1.300E-03
Case II.12	Sm-151	1.300E-05
Case II.13	Sm-152	2.100E-04
Case II.14	Eu-153	2.700E-04
Case II.15	Gd-155	3.600E-06
Case II.16	U-235	6.900E-04 (calibration U5: replace initial content)
Case II.17	U-236	2.400E-03
Case II.18	Np-237	4.400E-04
Case II.19	Pu-238	3.300E-04
Case II.20	Pu-239	4.900E-04
Case II.21	Pu-240	7.400E-05
Case II.22	Pu-241	2.600E-04
Case II.23	Pu-242	6.300E-04
Case II.24	Am-241	1.200E-04
Case II.25	Am-243	3.500E-04
Case II.26	B-10	2.700E-05 (calibration B10)
Case II.27	Sum of Cases I.1 to I.15 and I.17 to I.25	

In addition to BUC nuclides, air and light water are substituted to the central UO2 fuel to test the ability to predict the 'absolute' reactivity effect of UO_2 and to take into account scattering effects in the reactivity worth.

Materials to be substituted to the central fuel		Atomic density (atoms/barn.cm)	
Case II.28	Air	N-14	4.000E-05
		O-16	1.000E-05
Case II.29	Light Water	H-1	6.662E-02
		O-16	3.331E-02

3.3. Benchmark Phase III – 7x7 pattern cell calculation (axially finite)

Concentrations associated to the PWR fuel cells of the 7x7 lattice are identical to the ones of Phase I. Concentrations of the central cell materials are given below (temperature = 300K):

PWR cells	Same as Phase I	
Central UO_2 cell		
Fuel (UO_2 – 0.7%)	Same as Phase II	
Spacer	Atomic density (atoms/barn.cm)	
(AG-3)	Al	5.457E-02
	Mg	1.959E-03
Fuel-clad space	Same as PWR cells	
Clad	Same as PWR cells	
Moderator	Same as PWR cells	

Concentrations of the BUC nuclides to be added to the central UO_2 fuel are typical values encountered in small sample reactivity experiments, to provide reactivity worth of a few tens of pcm on the whole core (temperature = 300K):

Poisons to be added to the central fuel	
Case III.1 to III.27	Same as Phase II

In addition to BUC nuclides, air and light water are substituted to the central UO2 fuel to test the ability to predict the 'absolute' reactivity effect of UO_2 and to take into account scattering effects in the reactivity worth.

Materials to be substituted to the central fuel

Case III.28 to III.29	Same as Phase II

3.4. Benchmark Phase IV – 49x49 pattern cell calculation

Concentrations associated to the PWR fuel cells of the 49x49 lattice are identical to the ones of Phase I. Concentrations of the central cell materials are given below (temperature = 300K):

PWR cells	Same as Phase I

Central UO_2 cell

Fuel (UO_2 – 0.7%)	Same as Phase II
Spacer (AG-3)	Atomic density (atoms/barn.cm) Al 5.457E-02 Mg 1.959E-03
Fuel-clad space	Same as PWR cells
Clad	Same as PWR cells
Moderator	Same as PWR cells

Concentrations of the BUC nuclides to be added to the central UO_2 fuel are typical values encountered in small sample reactivity experiments, to provide reactivity worth of a few tens of pcm on the whole core (temperature = 300K):

Poisons to be added to the central fuel

Case III.1 to III.27	Same as Phase II

In addition to BUC nuclides, air and light water are substituted to the central UO2 fuel to test the ability to predict the 'absolute' reactivity effect of UO_2 and to take into account scattering effects in the reactivity worth.

Materials to be substituted to the central fuel

Case III.28 to III.29	Same as Phase II

4. Nuclear Data File

As the main objective of this benchmark is to compare numerical methods and models from different codes to emphasize specific problems associated with the analysis of reactivity-worth measurements, calculations should be performed with the same nuclear data library. It is recommended to use both the ENDF-VII.0 and the JEFF-3.1.1 libraries for this benchmark. Otherwise, JEFF-3.1.1 should be preferred to other libraries.

5. Case numbers

Each phase of the benchmark can be treated independently from the others. Case numbers are defined as follows:

5.1. Benchmark Phase I – single cell calculation

- Case I.0 represents the nominal case without any BUC nuclide (except U235 and U238) into the fuel.
- Cases I.1 to I.25 describe the addition of each BUC nuclide into the fuel following the concentrations given in §3.1.
- Case I.26 describes the addition of B-10 into the fuel as a calibration isotope for small-sample reactivity experiments with the concentration given in §3.1.

- Case I.27 describes the addition of all the 25 BUC nuclides (cases 1 to 25) into the fuel following the concentrations given in §3.1. For these two cases, the moderator, clad and gap for the central UO_2 fuel cell remain unchanged. This case corresponds to a PWR 40 GWd/t spent fuel lattice as modeled in BUC studies.

5.2. Benchmark Phase II – 7x7 pattern cell calculation (axially infinite)

- Case II.0 represents the nominal case without any BUC nuclide into the central fuel of the 7x7 pattern (except U235 and U238).
- Cases II.1 to II.25 describe the addition of BUC nuclides into the fuel following the concentrations given in §3.2.
- Case II.26 describes the addition of B-10 into the fuel as a calibration isotope for small-sample reactivity experiments with the concentration given in §3.2.
- Case II.27 describes the addition of all the 24 BUC nuclides (cases I.1 to I.15 and I.17 to I.25) into the fuel following the concentrations given in §3.1.
- Cases II.28 and II.29 describe the substitution of UO_2 fuel by respectively air and light water. For these two cases, the moderator, clad and gap for the central UO_2 fuel cell remain unchanged.

5.3. Benchmark Phase III – 7x7 pattern cell calculation (axially finite)

- Case III.0 represents the nominal case without any BUC nuclide into the central fuel of the 7x7 pattern (except U235 and U238).
- Cases III.1 to III.25 describe the addition of BUC nuclides into the fuel following the concentrations given in §3.3.
- Case III.26 describes the addition of B-10 into the fuel as a calibration isotope for small-sample reactivity experiments with the concentration given in §3.3.
- Case III.27 describes the addition of all the 24 BUC nuclides (cases I.1 to I.15 and I.17 to I.25) into the fuel following the concentrations given in §3.1.
- Cases II.28 and II.29 describe the substitution of UO_2 fuel by respectively air and light water. For these two cases, the moderator, clad and gap for the central UO_2 fuel cell remain unchanged.

5.4. Benchmark Phase IV – 49x49 pattern cell calculation

- Case IV.0 represents the nominal case without any BUC nuclide into the central fuel of the 49x49 pattern (except U235 and U238).
- Cases IV.1 to IV.25 describe the addition of BUC nuclides into the fuel following the concentrations given in §3.2.
- Case IV.26 describes the addition of B-10 into the fuel as a calibration isotope for small-sample reactivity experiments with the concentration given in §3.2.
- Case IV.27 describes the addition of all the 24 BUC nuclides (cases I.1 to I.15 and I.17 to I.25) into the fuel following the concentrations given in §3.1.
- Cases IV.28 and IV.29 describe the substitution of UO_2 fuel by respectively air and light water. For these two cases, the moderator, clad and gap for the central UO_2 fuel cell remain unchanged.

6. Parameters

Parameters of interest in this benchmark are given below. All integrations are defined over the central fuel volume (except for the denominator term in perturbation theory which is integrated over the whole geometry) and over 4π directions for solid angle. They are all normalized to the central fuel volume. The bra-ket notation <A, B> corresponds to the integration of the product A.B in the phase space, e.g. over energy, space and direction.

- Neutron flux $F = \int\int\int \Phi(E,\vec{r},\vec{\Omega})dEd\vec{r}d\vec{\Omega}$
- Absorption rate $A = \int\int\int \Sigma_a(E,\vec{r})\Phi(E,\vec{r},\vec{\Omega})dEd\vec{r}d\vec{\Omega}$
- Scattering rate (elastic and inelastic) $S = \int\int\int \Sigma_s(E,\vec{r})\Phi(E,\vec{r},\vec{\Omega})dEd\vec{r}d\vec{\Omega}$

- Production rate (normalisation to 1 source neutron)	$P=\iiint \nu\Sigma_f(E,\vec{r})\Phi(E,\vec{r},\vec{\Omega})dEd\vec{r}d\vec{\Omega}=k_{eff}$
- Reactivity worth (EigenValue-difference method)	$\rho_{EV}=\dfrac{k-k_o}{kk_o}$
- Reactivity worth (Exact Perturbation Theory)	$\rho_{EPT}=\dfrac{<\Phi_o^+,\delta H\Phi>}{<\Phi_o^+,P\Phi>}$
- Reactivity worth (Taylor Expansion Method)	ρ_{TEM}
- Reactivity worth (Multiple Estimation Method)	ρ_{MEM} (also referred as "correlated-sample theory")
- Capture effects in the reactivity worth	$\rho_C=\left.\dfrac{<\Phi_o^+,\delta H\Phi>}{<\Phi_o^+,P\Phi>}\right\rvert_{\Sigma_c}$
- Fission effects in the reactivity worth	$\rho_F=\left.\dfrac{<\Phi_o^+,\delta H\Phi>}{<\Phi_o^+,P\Phi>}\right\rvert_{\Sigma_f}$
- Scattering effects in the reactivity worth	$\rho_S=\left.\dfrac{<\Phi_o^+,\delta H\Phi>}{<\Phi_o^+,P\Phi>}\right\rvert_{\Sigma_s}$
- Leakage effects in the reactivity worth	$\rho_L=\left.\dfrac{<\Phi_o^+,\delta H\Phi>}{<\Phi_o^+,P\Phi>}\right\rvert_{leakage}$

Parameter definitions:

$\Phi(E,\vec{r},\vec{\Omega})$ is the perturbed neutron flux,

$\Phi_o^+(E,\vec{r},\vec{\Omega})$ is the unperturbed adjoint flux,

$\Sigma_a,\Sigma_c,\Sigma_f,\Sigma_s$ are respectively the macroscopic absorption, capture, fission and scattering cross sections,

ν is the number of neutrons emitted per fission,

k and k_o are the effective multiplication factor of respectively the perturbed case and unperturbated case.

L, *P* and *H* are respectively the Loss operator (including $\Sigma_a,\Sigma_f,\Sigma_s$ cross sections), the Production operator (including $\nu\Sigma_f$ cross section) and the Boltzmann operator (defined as $H = P / k - L$)

An accurate definition of ρ_{MEM} and ρ_{TEM} can be founded in the following paper:

> B. Morillon, *On the Use of Monte Carlo Perturbation in neutron transport problems*, Annals of Nuclear Energy, Volume 25, Issue 14, September 1998, Pages 1095-1117.

Guidances:

Each parameter is defined as "exactly" as possible. Nevertheless, the participants are invited to provide their results even in the case of simplified models, like for instance diffusion theory, anisotropy of scattering effects… Flux, Absorption Rates, Production Rates and Reactivity Worth from Eigenvalue difference method is the minimal set of results to be provided by the participants.

For users of Monte Carlo calculations, reactivity worth from Taylor Expansion of Multiple Estimate method should be given (if available).

For users of deterministic codes, reactivity worth from the Exact Perturbation Theory should be given (if available) including the decomposition between each cross section (at least capture and fission effects). Exact Perturbation Theory, which means using unperturbated adjoint flux and perturbated flux, corresponds to the real problem related to the measurement of reactivity-worth effects and should be preferred. Nevertheless, first-order perturbation theory could be provided if perturbated flux cannot be used in the perturbation analysis.

If leakage effects could not be explicitly calculated, a simplified expression could be used within the framework of the diffusion theory, leading to:

$\rho_L=\dfrac{\iint \nabla\Phi_o^+(E,\vec{r})\delta D(E)\nabla\Phi(E,\vec{r})dEd\vec{r}}{<\Phi_o^+,P\Phi>}$ where D is the diffusion coefficient

Another approximation could consist in the assumption of the fundamental mode (independency between energy and space) where flux and adjoint flux could be simplified like this: $\Phi(E,\vec{r},\vec{\Omega})=f(E)J_o(\alpha r)\cos(\beta z)$. In addition, in the heterogeneous P_1 model, the diffusion coefficient can be expressed as $D=1/(3\Sigma_{tr})$

7. Requested information and results

Please forward the results by electronic mail to CEA Cadarache (pierre.leconte@cea.fr) , one file per Phase of the benchmark. Results must be presented as follows:

Line No.	Contents	
1	*** BENCHMARK – PHASE I***	
2	Date	
3	Institute and Country	
4	Contact person	
5	E-mail address of the contact person	
6	Participants	
7	Computer Code	
8	Neutron data processing code or method	
9	Number of energy groups (NEG), supply 1 for continuous energy	
10	Upper energy Limit, in the sens from High to Low (i=1..NEG)	
11	Geometry modeling	
12	Omitted nuclides if any	
13	Numerical solver (Pij, MOC, Sn, Monte Carlo…)	
14	Employed convergence limit or statistical errors for eigenvalues	
15	* Results of Case I.0 *	
16	Multiplication Factor (for Monte Carlo: Number of histories, Deviation)	
17	- Reaction rates – (Total over all energy groups)	
18	Neutron flux	*F*
19	U-235 absorption rate	*A*
20	U-235 production rate	*P*
21	U-235 scattering rate	*S*
22	U-238 absorption rate	*A*
23	U-238 production rate	*P*
24	U-238 scattering rate	*S*
25	O-16 absorption rate	*A*
26	O-16 scattering rate	*S*
27	* Results of Case I.1 *	
28	Multiplication Factor (for Monte Carlo: Number of histories, Deviation)	
29	- Reaction rates – (Total over all energy groups)	
30	Neutron flux	*F*
31	U-235 absorption rate	*A*
32	U-235 production rate	*P*
33	U-235 scattering rate	*S*
34	U-238 absorption rate	*A*
35	U-238 production rate	*P*
36	U-238 scattering rate	*S*
37	O-16 absorption rate	*A*
38	O-16 scattering rate	*S*
39	Added isotope absorption rate	*A*
40	Added isotope production rate	*P*
41	Added isotope scattering rate	*S*
42	Total reactivity variation	ρ_{EV}
43	Total reactivity variation from Perturbation Theory	ρ_{EPT} (for a deterministic calculation) ρ_{TEM} or ρ_{MEM} (for a probabilistic calculation)
44	Reactivity variation due to capture	ρ_C
45	Reactivity variation due to fission	ρ_F
46	Reactivity variation due to scattering	ρ_S
47	Reactivity variation due to leakage	ρ_L

Lines 27 to 47 are repeated for cases 2 to 28 (or 26 for Phase I).

If participants are not able to provide the full set of results, they can send a reduced file leaving blank spaces for parameters which cannot be obtained. The minimal set should contain results for:

- lines 18, 19, 20, 22, 23, 30, 31, 32, 34, 35, 39, 40, 42 for a deterministic calculation (with results in lines 43, 44 and 45 if perturbation theory is available)
- lines 18, 19, 20, 22, 23, 30, 31, 32, 34, 35, 39, 40, 42 for a probabilistic calculations (with results in lines 43, 44 and 45 if perturbation theory is available)

Participants who have the ability to perform both deterministic and probabilistic calculations should send their results in separated files.

8. Schedule

The following schedule is proposed:

- Mid-September 2010
  Distribution of the draft specification
- Mid-December 2010
  Comments on available nuclides and parameters
- End December 2010
  Distribution of final specification
- End June 2011
  Final results on Phase I and II from WPNCS/BUC participants to CEA
- September 2011
  Presentation of the results of Phase I and II at the WPNCS meeting
- End June 2012
  Final results on Phase III and IV from WPNCS/BUC participants to CEA
- September 2012
  Presentation of all the results of Phase I to IV at the WPNCS meeting

Fig. 1: Part 1 - PWR cell (radial cross section)

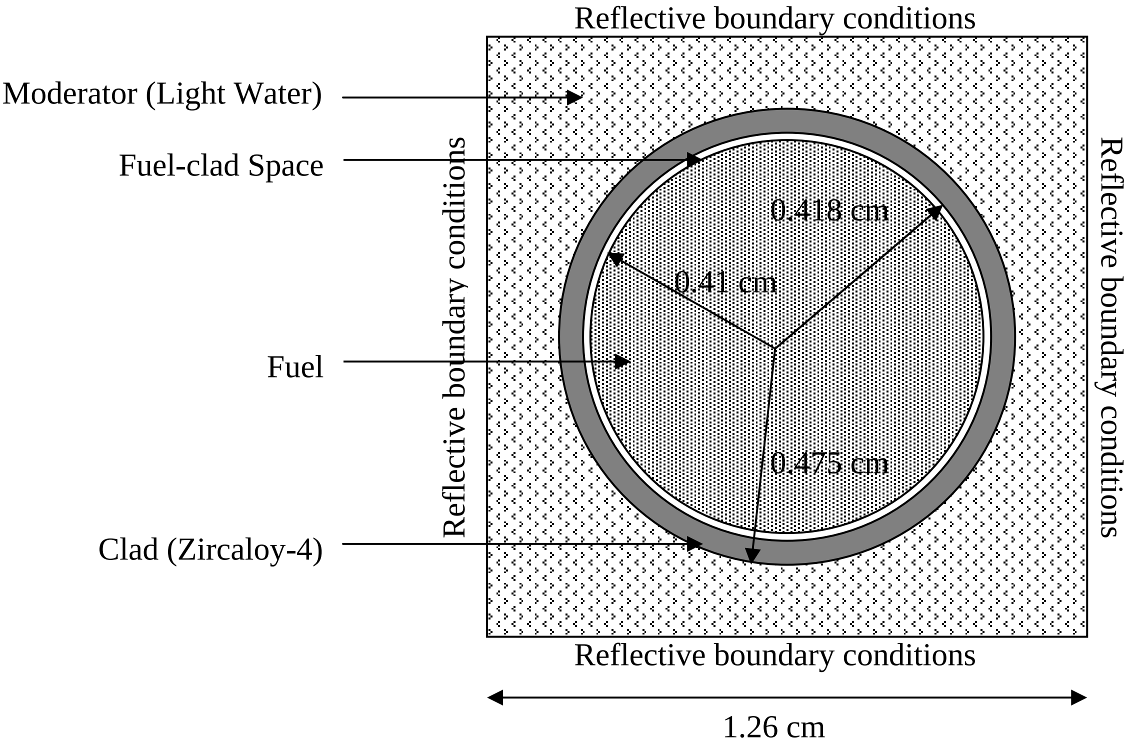

Fig. 2 : Part 2 – Axially infinite 7x7 lattice (radial cross section)

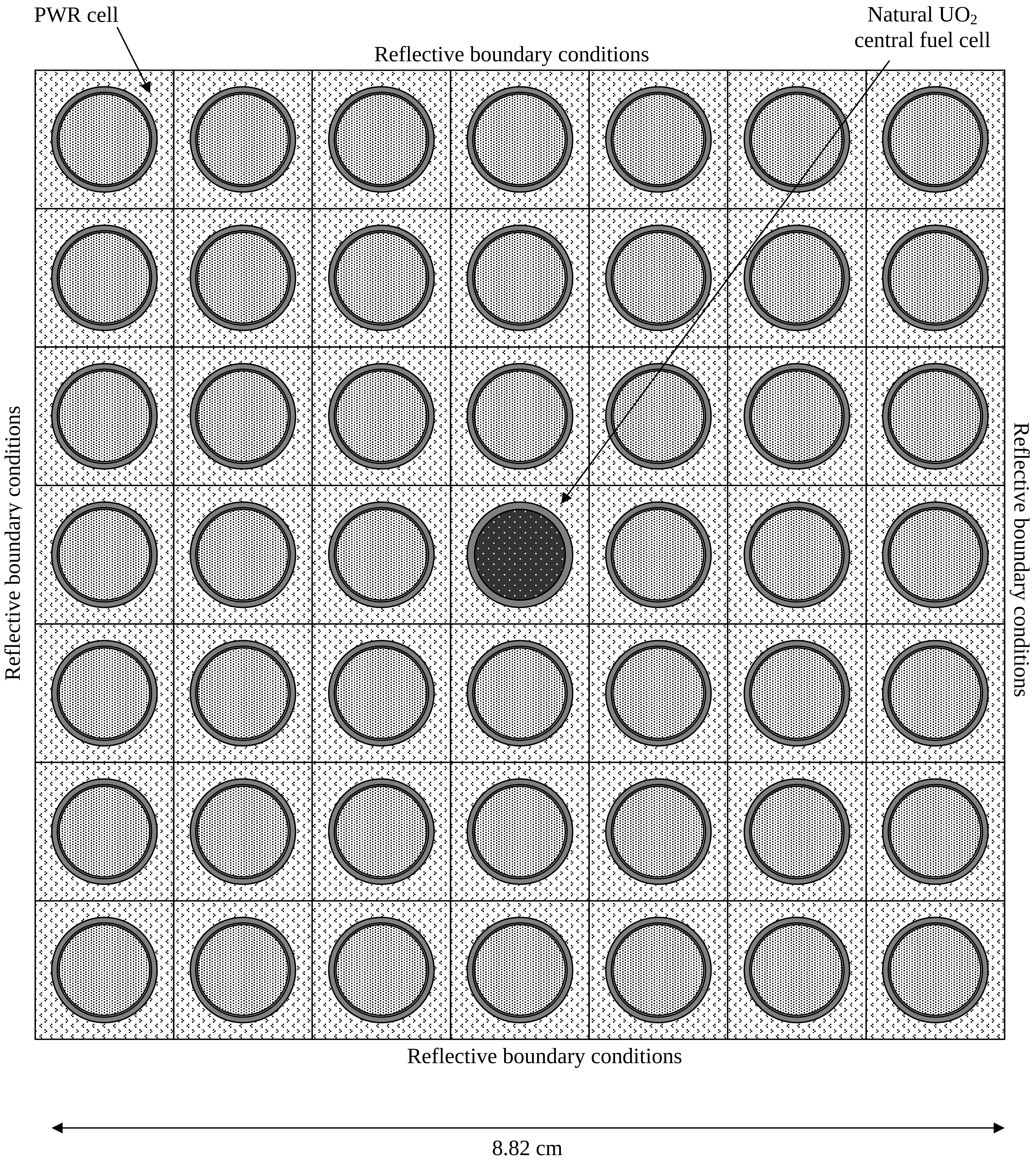

Fig. 3 : Part 3 – Axially finite 7x7 lattice (axial cross section, non to scale)

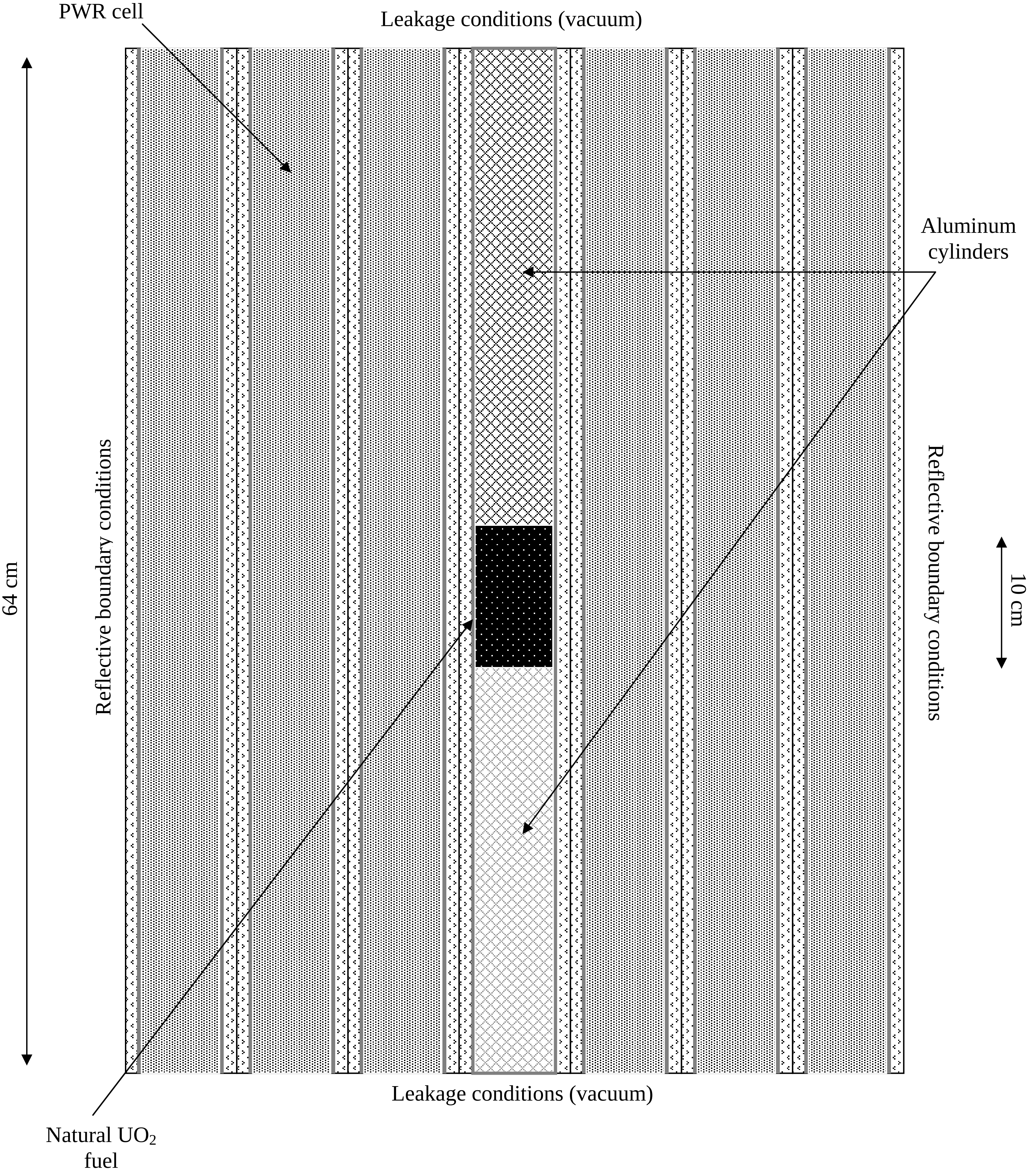

Fig. 4 : Part 4 – axially and radially finite 49x49 lattice (radial cross section)