Appendix A

Benchmark specification for plutonium recycling in PWRs

Benchmark A: Poor-quality plutonium
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Benchmark B: Better plutonium vector
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Benchmark A – poor-isotopic-quality plutonium

The goal of this comparison is to explain the reasons for unexplained differences between results on MOX-PWR cell calculations using degraded plutonium (fifth-stage recycle).

The most important difference is related to the infinite medium multiplication constant k-infinity. We suggest a geometry as simple as possible. We shall describe the proposed options:

- Number of atoms and cell geometry

  Differences could appear for these calculations. So we propose that a number of atoms will be stated for the benchmark.

  For this preliminary calculation, we have taken the geometry of Figure A-1 and the following isotopic balance of plutonium. The plutonium isotopic composition is near the composition at the fifth stage recycle with an average burnup of 50 MWd/kg.

Pu-238	4%
Pu-239	36%
Pu-240	28%
Pu-241	12%
Pu-242	20%

The uranium isotopic composition is the following

U-235	0.711%
U-238	99.289%

The total plutonium concentration proposed is 12.5% (6% of fissile plutonium).

The cladding is only made out of natural zirconium.

In evolution, samarium and xenon concentrations will be self-estimated by each code with a nominal power of 38.3 W/g of initial heavy metal.

- Options of the cell calculation

To ease the comparisons, it is suggested to calculate the cell without any neutron leakage ($B^2 = 0$).

Temperatures will be as follows:

- Fuel 660°C
- Cladding 306.3°C
- Water 306.3°C

Boron concentration is worth 500 ppm. Boron composition is as follows:

- B-10 18.3%
- B-11 81.7%

FUEL	
	ATOMS / cm^3
U-234	0
U-235	$1.4456 \cdot 10^{20}$
U-236	0
U-238	$1.9939 \cdot 10^{22}$
Np-237	0
Pu-238	$1.1467 \cdot 10^{20}$
Pu-239	$1.0285 \cdot 10^{21}$
Pu-240	$7.9657 \cdot 10^{20}$
Pu-241	$3.3997 \cdot 10^{20}$
Pu-242	$5.6388 \cdot 10^{20}$
Am-241	0
Am-242	0
Am-243	0
Cm-242	0
Cm-243	0
CLADDING	
natural Zr	$4.5854 \cdot 10^{22}$
MODERATOR	
H_2O	$2.3858 \cdot 10^{22}$
B-10	$3.6346 \cdot 10^{18}$
B-11	$1.6226 \cdot 10^{19}$

Table A-1 ***Number of atoms per cm^3 at irradiation step zero***

- Options of the evolution calculation

We propose an evolution calculation from 0 to 50 MWd/kg including the following time steps (0, 0.15, 0.5, 1, 2, 4, 6, 10, 15, 20, 22, 26, 30, 33, 38, 42, 47 and 50 MWd/kg)

We take into consideration the following fission products:

Zr-95, Mo-95, Pd-106, Ce-144, Pm-147, Pm-148, Pm-148m, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Eu-154, Eu-155, Gd-155, Gd-156, Gd-157, Tc-99, Ag-109, Cd-113, In-115, I-129, Xe-131, Cs-131, Cs-137, Nd-143, Nd-145, Nd-148
and four pseudo fission products in which all the other fission products are grouped.

The energy releases from fission are:

NUCLIDE	ENERGY RELEASE (MeV)
U-235	193.7
U-238	197.0
Pu-239	202.0
Pu-241	204.4
Am-242m	207.0

plus 8 MeV for the n-gamma captures of the other non-fissioning (ν-1) neutrons.

- Results

Results should be provided both on paper and computer-processable medium. A short report should be provided describing:

- The computer program(s) used and their precise version,
- The data libraries used and evaluated data file from which they were derived,
- The list of isotopes for which resonance self-shielding was applied and the method used,
- How the buildup of Xenon was treated,
- How the (n,2n)-reaction was taken into account for the k-infinity calculation.

The following data should be provided in tabular form for the following burnups:
0, 10, 33, 42 and 50 MWd/kg.

1. Number densities for all nuclides considered:

	burnup 1	burnup 2	....	burnup-n
isotope 1				
isotope 2				
.				
.				
.				
.				
.				
isotope -N				

2. k as a function of burnup,

3. One energy group cross-section (absorption, fission, nu-bar) as a function of isotope and burnup (see 1.),

4. Reaction rates (absorption, fission) as a function of isotope and burnup (see 1.),

5. Applied absolute fluxes used in the evolution calculation (and their normalisation factor),

6. Neutron energy spectrum per unit lethargy as a function of burnup (and its normalisation factor and group structure).

Benchmark B – better plutonium vector

As a second fuel M2, in agreement both with Dr. G. Schlosser, KWU and Dr. J. Vergnes, EDF, a MOX fuel with first-generation-plutonium as used in [1] with the following specifications is suggested:

- 4.0 wt% U-235 in uranium tailings (0.25 wt% U-235),
- Composition of plutonium (wt%):

Pu-238	1.8
Pu-239	59.0
Pu-240	23.0
Pu-241	12.2
Pu-242	4.0

- Composition of uranium (wt%):

U-234	0.00119
U-235	0.25
U-238	99.74881

With the heavy material number density normalised to 2.115×10^{22} atoms $/cm^3$, the following nuclide number densities are determined:

NUCLIDE	ATOMS / cm^3
U-234	$2.4626 \cdot 10^{17}$
U-235	$5.1515 \cdot 10^{19}$
U-238	$2.0295 \cdot 10^{22}$
Pu-238	$2.1800 \cdot 10^{19}$
Pu-239	$7.1155 \cdot 10^{20}$
Pu-240	$2.7623 \cdot 10^{20}$
Pu-241	$1.4591 \cdot 10^{20}$
Pu-242	$4.7643 \cdot 10^{19}$
heavy metal-atoms	$2.155 \cdot 10^{22}$
O	$4.310 \cdot 10^{22}$

All other specifications shall be the same as in the first benchmark – case A.

Reference

[1] H. W. Wiese, "Investigation of the Nuclear Inventories of High-Exposure PWR Mixed Oxide Fuels with Multiple Recycling of Self-Generating Plutonium", Nuclear Technology, Vol. 102, April 1993, p. 68.

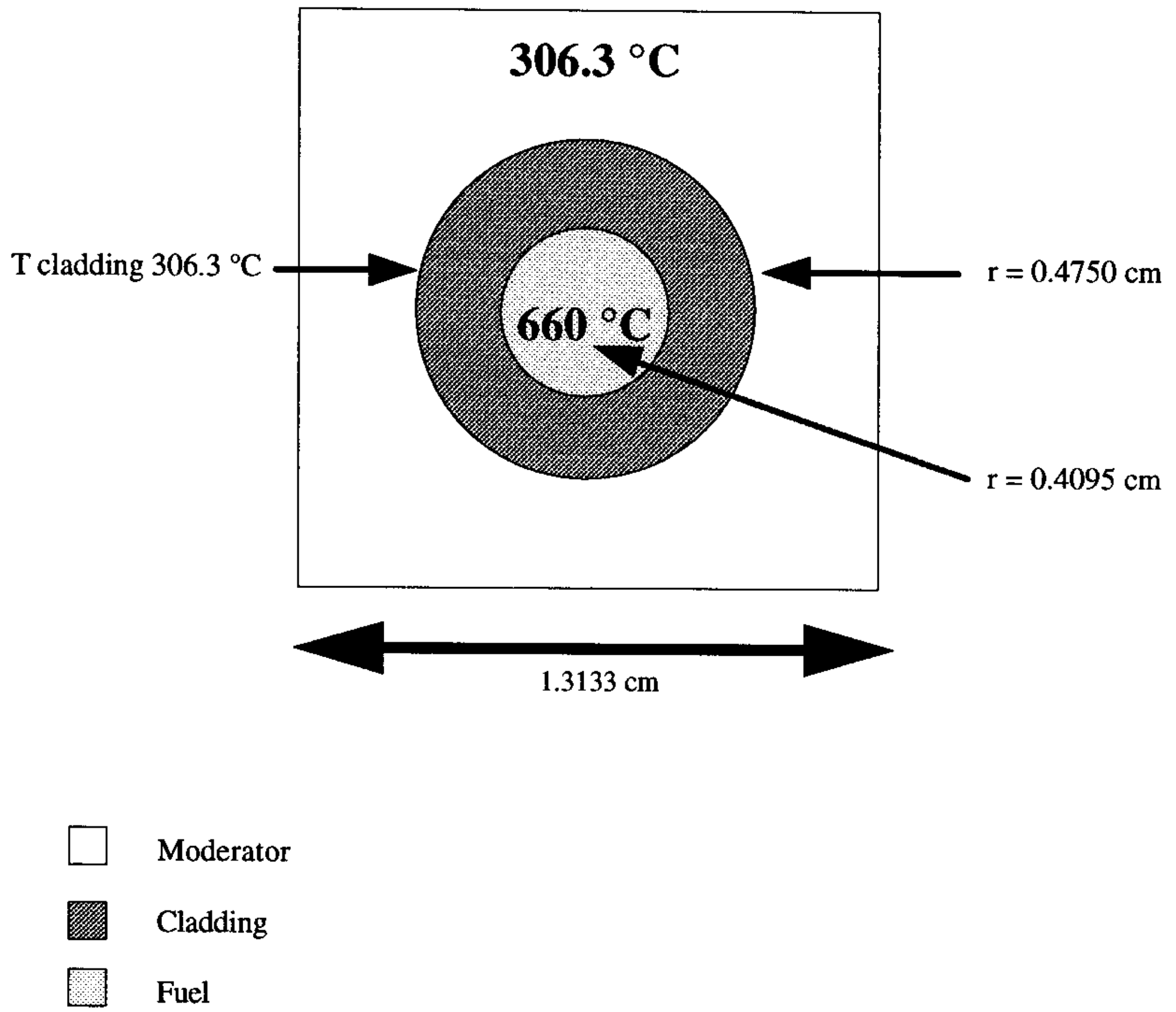

Figure A-1 Cell geometry at 20°C