

# **Fuel cycle parameters from SG26: Choices and volunteers**

**M.Salvatores**

**SG46 Meeting 25-26 November 2019**

# SG26

The parameters considered were:

- Criticality (multiplication factor);
- Power peak value;
- Doppler reactivity coefficient;
- Coolant void reactivity coefficient;
- Reactivity loss during irradiation;
- Transmutation potential (i.e. nuclide concentrations at end of irradiation,  $n_f$ , or density variation during burnup,  $\Delta n = n_f - n_i$ );
- Decay heat in a repository (at 100 years after disposal);
- Radiotoxicity at  $t=100000$  years after disposal.
- Radiation source at fuel discharge;

# SG26: Examples of nominal values

ABTR, SFR, EFR Nominal Values

Reactor	Reactivity [pcm]	Power Peak	Doppler [pcm]	Void [pcm]	Burnup [pcm]	Decay Heat [W]	Dose [Sv]	Neutron Source [n/s]
<b>ABTR</b>	3319.8	1.73 <sup>(a)</sup>	368.3 <sup>(b)</sup>	277 <sup>(c)</sup>	-1325.8 <sup>(d)</sup>	1.402E+03	1.687E+01	7.005E+09
<b>SFR</b>	5015.4	1.53 <sup>(e)</sup>	231 <sup>(f)</sup>	1837 <sup>(g)</sup>	-3981.1 <sup>(h)</sup>	1.400E+4	2.319E+1	8.537E+11
<b>EFR</b>	9786.5	1.63 <sup>(i)</sup>	1289 <sup>(l)</sup>	1934.5 <sup>(m)</sup>	-9123.9 <sup>(n)</sup>	4.324E+3	1.235E+1	1.996E+11

<sup>(a)</sup> (R , Z) =(10.6 , 151.3)<sub>cm</sub>

<sup>(b)</sup> T<sub>fuel</sub>=300K - T<sub>fuel</sub>=856K

<sup>(c)</sup> Na loss at the core center

<sup>(d)</sup> 109.8 days (4 months at 90% capacity)

<sup>(e)</sup> (R , Z) =(66.59 , 143.03)<sub>cm</sub>

<sup>(f)</sup> T<sub>fuel</sub>=300K - T<sub>fuel</sub>=850K

<sup>(g)</sup> Na loss in core

<sup>(h)</sup> 155 days

<sup>(i)</sup> (R , Z) =(153.24 , 125)<sub>cm</sub>

<sup>(l)</sup> T<sub>fuel</sub>=300K - T<sub>fuel</sub>=1520K

<sup>(m)</sup> Na loss in core and blanket

<sup>(n)</sup> 1700 days

<sup>(o)</sup> 100 y after discharge

<sup>(p)</sup> 100000 y after dischar

<sup>(q)</sup> 2 y after discharge

LFR Nominal Values

Reactivity [pcm]	Power Peak	Doppler [pcm]	Void [pcm]	Burnup [pcm]	Decay Heat [W]	Dose [Sv]	Neutron Source [n/s]
22.9	1.29 <sup>(a)</sup>	228.1 <sup>(b)</sup>	6575.5 <sup>(c)</sup>	-1464 <sup>(d)</sup>	4.616E+3	1.140E+1	2.275E+11

<sup>(a)</sup> (R , Z) =(100.96 , 117.90)<sub>cm</sub>

<sup>(b)</sup> T<sub>fuel</sub>=300K - T<sub>fuel</sub>=900K

<sup>(c)</sup> Pb loss in core

<sup>(d)</sup> 310 days

Table 1. ADMAB Nominal Values

Reactivity [pcm]	Power Peak	Doppler [pcm]	Void [pcm]	Burnup [pcm]	Decay Heat [W]	Dose [Sv]	Neutron Source [n/s]
-5466.9	2.67 <sup>(a)</sup>	28.3 <sup>(b)</sup>	3138.4 <sup>(c)</sup>	-1347.6 <sup>(d)</sup>	1.545E+5	2.653E+1	3.122E+12

<sup>(a)</sup> (R , Z) =(20 , 102.5)<sub>cm</sub>

<sup>(b)</sup> T<sub>fuel</sub>=1773K - T<sub>fuel</sub>=980K

<sup>(c)</sup> Pb-Bi loss in core

<sup>(d)</sup> 366 days



# SG26: Examples of uncertainties

Fast Neutron Systems: Total Uncertainties (%)

Reactor		$k_{eff}$	Power Peak	Doppler	Void	Burnup [pcm]	Decay Heat	Dose	Neutron Source
ABTR	BOLNA <sup>(b)</sup>	0.92	0.3	4.4	6.0	52	0.2	0.1	0.5
SFR	BOLNA	1.82	0.4	5.6	17.1	272	0.4	0.3	1.0
EFR	BOLNA	1.18	1.2	3.8	7.8	871	2.4	1.2	6.6
GFR	BOLNA	1.88	1.7	5.5	7.7	381	0.4	0.5	1.4
LFR	BOLNA	1.43	0.6	4.3	7.2	198	0.6	0.4	1.1
ADMAB	BOLNA	2.94	21.4	-	15.5	1044	0.7	1.0	2.5

<sup>(a)</sup> Partial Energy Correlation as used in Ref. 1

<sup>(b)</sup> BNL\_ORNL\_LANL\_NRG\_ANL

High Burnup VHTR: Uncertainties (%)

	$k_{eff}$ BOC	$k_{eff}$ EOC	Power Peak BOC	Power Peak EOC	Doppler BOC	Doppler EOC	Burnup [pcm]	Decay Heat	Dose	Neu Sou
BOLNA <sup>(b)</sup>	0.53	0.46	1.0	1.1	1.7	2.0	530	1.4	1.0	5.

<sup>(a)</sup> Partial Energy Correlation as used in Ref. 1

<sup>(b)</sup> BNL\_ORNL\_LANL\_NRG\_ANL

High Burnup PWR: Uncertainties (%)

	$k_{eff}$ BOC	$k_{eff}$ EOC	Doppler BOC	Doppler EOC	Burnup [pcm]	Decay Heat	Dose	Neutron Source
BOLNA <sup>(b)</sup>	0.51	0.74	1.4	1.9	851	1.5	1.0	5.2

Uncertainty (%) on Pu Isotope Density at End of Cycle (Case of EFR)

Uncertainty on →		Pu238	Pu239	Pu240	Pu241	Pu242
↓ due to						
U238	capture	-	1.1	0.2	0.1	-
Pu238	capture	1.7	0.1	-	-	-
	fission	4.6	-	-	-	-
Pu239	capture	-	0.8	1.3	0.7	0.1
	fission	-	0.2	-	-	-
Pu240	capture	0.2	-	1.5	6.0	1.0
	fission	-	-	0.8	0.4	-
Pu241	capture	-	-	-	0.8	1.5
	fission	0.2	-	-	5.0	0.7
Pu242	capture	-	-	-	-	3.9
	fission	-	-	-	-	2.2
Am241	capture	1.3	-	-	-	0.2
	fission	0.2	-	-	-	-
<b>Total</b>		<b>5.1</b>	<b>1.3</b>	<b>2.1</b>	<b>7.9</b>	<b>4.9</b>

Uncertainty (%) on Selected MA Density at End of Cycle (Case of EFR)

Uncertainty on →		Am241	Am242m	Am243	Cm244	Cm245
↓ due to						
Pu240	capture	1.6	0.6	0.2	-	-
	fission	0.1	-	-	-	-
Pu241	capture	0.2	0.1	0.4	0.1	-
	fission	1.2	0.4	0.1	-	-
Pu242	capture	-	-	9.3	4.1	1.5
	fission	-	-	0.6	0.2	-
Am241	capture	3.1	2.0	0.1	-	-
	fission	0.9	0.5	-	-	-
Am242m	capture	-	1.6	0.3	0.2	0.1
	fission	-	7.4	0.1	-	-
Am243	capture	-	-	1.9	1.9	1.0
	fission	-	-	0.5	0.2	0.1
Cm244	capture	-	-	-	1.8	7.2
	fission	-	-	-	6.0	2.8
Cm245	capture	-	-	-	-	0.9
	fission	-	-	-	-	15.6
<b>Total</b>		<b>3.8</b>	<b>7.8</b>	<b>9.5</b>	<b>7.8</b>	<b>17.6</b>

As for **data needs related to fuel cycle parameters**, the present study has addressed in a quantitative way the uncertainties of a number of parameters directly related to the nuclei density uncertainties. A more comprehensive study related, e.g., to the decay heat, is still to be done. However, the Subgroup has addressed in a preliminary way several important issues. In particular **Appendix P** deals with the case of **nuclear data for the handling, reprocessing and disposal of spent nuclear fuel**. Since it is recognized that for future facilities any nuclear data which dominates the uncertainties need to be identified as well as any biases so that improvements can be made to the basic nuclear data, an initial study is described in **Appendix P, considering only thermal reactors** and some conclusions made about how these could be extended for novel systems. As an example, the calculation of decay heat in the context of the current fuel reprocessing plants is discussed there.

It can be concluded that decay heat from LWR spent fuel can be adequately determined using existing codes and data. **Although nuclides can be identified where improvements may be possible, this will have little benefit to existing operations, although it may assist future technology development.**

However, as **no such measurements are published for spent fuels from fast reactors and ADS systems**, this analysis cannot be repeated for these systems. Thus biases and uncertainties on their decay heat cannot be determined by this method (**R. Mills**).