Nuclear Data Needs for Advanced Reactors and Fuel Cycles

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Advanced Reactors

- **In general, the uncertainty analysis performed using current covariance data shows that the present integral parameters uncertainties resulting from the assumed uncertainties on nuclear data are probably acceptable in the early phases of design feasibility studies.**
- **However, in the successive phase of preliminary conceptual designs and in later design phases of selected reactor and fuel cycle concepts, there is the need for improved data and methods, in order to reduce margins, both for economic and safety reasons.**
- **It is then important to define as soon as possible priority issues, i.e. which are the nuclear data (isotope, reaction type, energy range) that need improvement, in order to quantify target accuracies and to select a strategy to meet the requirements needed (e.g. by some selected new differential measurements and by the use of integral experiments).**

Background

- **The ultimate goal is a design that has as low as possible uncertainties. Industry and utilities want reduced uncertainty for economical reasons (design and operation), while safety authorities want “guaranteed margins” that they can trust.**
- **There are two main sources of uncertainties: input data, and modeling**
  - **Example of input physical data: cross sections, fabrication data, etc.**
  - **Modeling uncertainties: coming from approximations made in the computational methodology used in the design process.**
- **High-fidelity simulation can provide a major benefit if it can reduce to the smallest amount the impact of uncertainties coming from the modeling of the physical processes.**
- **A scientific based approach can allow a reliable propagation of uncertainties and a correct evaluation of the impact of the uncertainty coming from the input data.**

Sample of Target Reactor Performance (Fast Reactor)

Parameter	Current Uncertainty (SFR)		Targeted Uncertainty
	Input data origin (a priori)	Modeling origin	
Multiplication factor, K_{eff} ($\Delta k/k$)	1.5%	0.5%	0.3%
Power peak	1%	3%	2%
Power distribution[d)]	1%	6%	3%
Conversion ratio (absolute value in %)	5%	2%	2%
Control rod worth: Element	5%	6%	5%
Control rod worth: Total	5%	4%	2%
Burnup reactivity swing ($\Delta k/k$)	0.7%	0.5%	0.3%

Uncertainty Evaluation

- **Sensitivity and uncertainty analyses are the main instruments for dealing with the sometimes scarce knowledge of the input parameters used in simulation tools.**
- **Sensitivity coefficients are the key quantities that have to be evaluated. They are determined and assembled, using different methodologies, in a way that when multiplied by the variation of the corresponding input parameter they will quantify the impact on the targeted quantities whose sensitivity is referred to.**

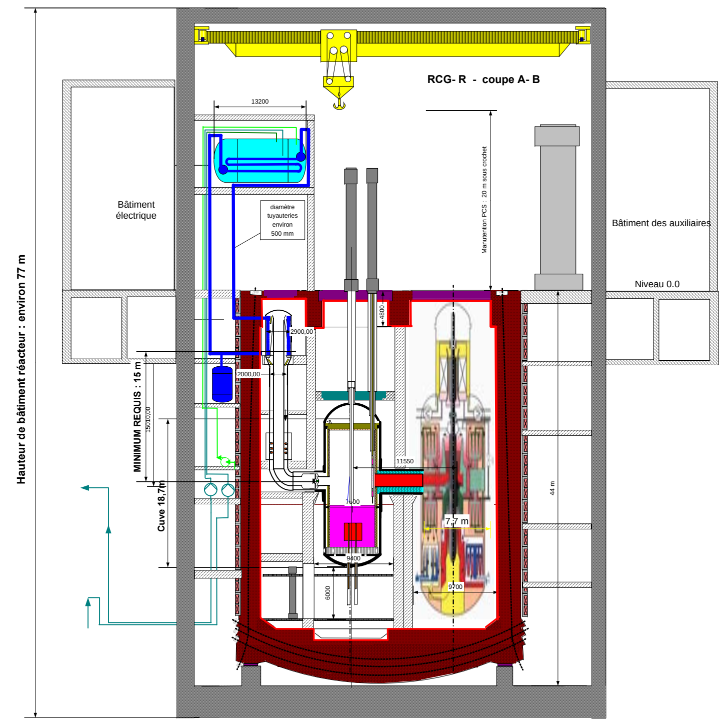
$$\Delta \mathbf{R}^2 = \mathbf{S}_\mathbf{R}^+ \mathbf{D} \mathbf{S}_\mathbf{R}$$

where ΔR is the uncertainty, S_R are the sensitivity coefficients arrays, and D is the covariance matrix.

Nuclear data can have a significant impact on innovative design features

- A wide range of systems has been investigated, both within the AFCI and GEN-IV programs
- Some expected new significant features (core and fuel cycle) depend heavily on nuclear data knowledge and uncertainties.
- Typical examples of nuclear data dependent innovative design features are:
  - Cores with low reactivity loss during the cycle
  - Cores with increased inventory of Minor Actinides in the fuel
  - Cores with no uranium blankets
- Both core design and the associated fuel cycle features have to be considered

Fast Reactors



RCG-R Cavité béton précontraint coupe A-A

Breeder or Burner

Fuels, structures, and reflectors might contain new materials (Zr, Si)

Uncertainties and Target Accuracies: Lessons Learned with WPEC Subgroup 26

Recent work to assess uncertainties on a wide range of integral parameters and a wide range of systems, has been performed within an international initiative and a final report has been issued :

"OECD/NEA WPEC Subgroup 26 Final Report: Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations"

This work has been made possible by the work on covariance data, led by BNL with LANL and ORNL participation (the so-called BOLNA covariance data set) , and by the availability of state-of-the-art sensitivity analysis tools

SFR (Burner: CR = 0.25)

840 MW_{th} – Na Cooled

U-TRU-Zr Metallic Alloy Fuel

SS Reflector

Pu content: 56%

MA: 10%

Irradiation Cycle: 155 d

EFR

3600 MW_{th} – Na Cooled

U-TRU Oxide Fuel

U - Blanket

Pu content : 22.7%

MA: 1%

Irradiation Cycle: 1700 d

GFR

2400 MWe – He Cooled

SiC – (U-TRU)C Fuel

Zr_3Si_2 Reflector

Pu content : 17%

MA: 5%

Irradiation Cycle: 415 d

LFR

900 MW_{th} – Pb Cooled

U-TRU-Zr Metallic Alloy Fuel

Pb Reflector

Pu content : 21%

MA: 2%

Irradiation Cycle: 310 d

VHTR

TRISO Fuel

U235 Enrichment: 14%

Burnup: 90 GWd/Kg

Some of the systems which have been investigated

Nuclear data uncertainties have impact on :

Reactor parameters….

- **Criticality (multiplication factor)**
- **Doppler Reactivity Coefficient**
- **Coolant Void Reactivity Coefficient**
- **Reactivity Loss during Irradiation**
- **Transmutation Potential (i.e. nuclide concentration at the end of irradiation)**
- **Peak Power Value**
- **Etc**
- **….and fuel cycle parameters:**

- **MA Decay Heat in a Repository**
- **Radiation Source at Fuel Discharge**
- **Radiotoxicity in a Repository**
- **Etc**

SFR Uncertainties (%) - Breakdown by Isotope (Diagonal Values)

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.16	0.05	0.60	1.65	10.5
Pu238	0.34	0.01	0.86	2.72	45.6
Pu239	0.13	0.02	0.49	1.39	20.6
Pu240	0.38	0.03	0.96	3.83	32.2
Pu241	0.52	0.02	1.70	4.34	89.8
Pu242	0.26	0.02	0.74	2.65	24.4
Np237	0.03	0.01	0.23	0.40	1.2
Am241	0.07	0.01	0.34	0.62	3.4
Am242m	0.37	0.02	1.08	3.06	50.4
Am243	0.05	0.01	0.31	0.53	5.8
Cm242	0.02	-	0.06	0.14	8.6
Cm243	0.01	-	0.02	0.05	2.3
Cm244	0.27	0.01	0.66	2.84	42.6
Cm245	0.19	0.01	0.49	1.28	31.5
Fe56	0.37	0.13	1.89	4.44	31.4
Cr52	0.04	0.01	0.27	0.47	2.2
Zr90	0.03	0.02	0.10	0.24	2.3
Na23	0.23	0.10	1.25	12.29	19.6
B10	0.12	0.24	0.22	1.16	8.7
Total	1.04	0.31	3.62	15.66	152.1

SFR Uncertainties (%) - Breakdown by Isotope (With Corr.)

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.24	0.07	0.94	2.43	16.0
Pu238	0.64	0.02	1.50	3.00	83.2
Pu239	0.19	0.04	0.71	1.75	29.3
Pu240	0.66	0.05	1.60	3.86	56.9
Pu241	0.96	0.02	2.77	4.12	170.2
Pu242	0.41	0.03	1.15	3.37	37.5
Np237	0.06	0.01	0.31	0.51	2.1
Am241	0.11	0.01	0.55	0.91	5.6
Am242m	0.73	0.02	1.84	3.73	100.7
Am243	0.07	0.01	0.49	0.78	8.8
Cm242	0.04	-	0.10	0.13	15.5
Cm243	0.02	-	0.04	0.03	4.5
Cm244	0.40	0.02	1.00	3.01	64.5
Cm245	0.39	0.01	0.95	1.00	62.2
Cm246	0.04	-	0.14	0.28	4.1
Fe56	0.55	0.20	2.48	4.47	47.0
Cr52	0.06	0.01	0.38	0.51	2.9
Zr90	0.03	0.03	0.12	0.29	2.5
Na23	0.25	0.13	1.85	13.53	21.6
B10	0.17	0.36	0.35	1.53	12.8
Total	1.82	0.45	5.57	17.11	271.9

SFR K_{eff} Uncertainties (%) – Energy Breakdown for Selected Isotope/Reaction

Group	Energy	Pu-238 $\sigma_{fission}$	Pu-240 $\sigma_{capture}$	Pu-241 $\sigma_{fission}$	Am-242m $\sigma_{fission}$
1	19.6 MeV	0.01	0.00	0.02	0.02
2	6.07 MeV	0.18	0.03	0.10	0.12
3	2.23 MeV	0.23	0.05	0.26	0.15
4	1.35 MeV	0.31	0.11	0.40	0.28
5	498 keV	0.28	0.14	0.47	0.39
6	183 keV	0.12	0.16	0.58	0.39
7	67.4 keV	0.07	0.13	0.29	0.28
8	24.8 keV	0.06	0.13	0.16	0.12
9	9.12 keV	0.03	0.05	0.10	0.08
10	2.03 keV	0.03	0.01	0.08	0.10
11	454 eV	0.00	0.00	0.03	0.02
12-15	22.6 eV	0.00	0.00	0.00	0.00
Total		0.53	0.31	0.96	0.73

Target Accuracy Requirements

Target accuracy assessments are the inverse problem of the uncertainty evaluation. To establish priorities and target accuracies on data uncertainty reduction, a formal approach can be adopted by defining target accuracy on design parameter and finding out required accuracy on data. In fact, the unknown uncertainty data requirements can be obtained by solving a minimization problem where the sensitivity coefficients in conjunction with the existing constraints provide the needed quantities to find the solutions.

$$Q = \sum_i \lambda_i / d_i^2 = min \quad i = 1 \ldots I$$

with the following constraints:

$$\sum_i S_{ni}^2 d_i^2 + \sum_{ii'} S_{ni}^n d_i Corr_{ii'} d_{i'} S_{ni'} \leq \left(R_n^T\right)^2 \quad n = 1..N$$

where *N* is the total number of integral design parameters, S_{ni} are the sensitivity coefficients for the integral parameter R_n and are the required target accuracies on the *N* integral parameters; λ_i are “cost” parameters related to each σ_i and should give a relative figure of merit of the difficulty of improving that parameter (e.g., reducing uncertainties with an appropriate experiment), and $Corr_{ii'}$ are the correlation values between variable *i* and *i’*.

Target Accuracy Assessment

Targeted Accuracies for Main design Parameters

Multiplication factor (BOL)	300 pcm
Power peak (BOL)	2%
Burnup reactivity swing	300 pcm
Reactivity coefficients (Coolant void and Doppler - BOL)	7%
Major nuclide density at end of irradiation cycle	2%
Other nuclide density at end of irradiation cycle	10%

Target Accuracy Assessment: ABTR

Isotope	Cross-Section	Energy Range	Uncertainty (%) Initial	Required $\lambda=1$	Required $\lambda\neq1$ case B [(a)]
U238	σ_{inel}	6.07 - 2.23 MeV	19.8	3.3	5.8
		2.23 - 1.35 MeV	20.6	3.6	6.3
		1.35 - 0.498 MeV	11.6	6.5	11.4
U238	σ_{capt}	24.8 - 9.12 keV	9.4	2.9	1.6
Pu239	σ_{capt}	498 - 183 keV	11.6	5.7	3.2
		183 - 67.4 keV	9.0	5.0	2.8
		67.4 - 24.8 keV	10.1	5.8	3.2
		9.12 - 2.04 keV	15.5	7.4	4.1
Pu241	σ_{fiss}	183 - 67.4 keV	19.9	8.8	7.0
Fe56	σ_{inel}	2.23 - 1.35 MeV	25.4	5.6	9.9
		1.35 - 0.498 MeV	16.1	7.5	13.1
Na23	σ_{inel}	1.35 - 0.498 MeV	28.0	10.1	17.7

	$\lambda=1$	$\lambda\neq1$ case A	$\lambda\neq1$ case B
$\lambda_{capt,fiss,\nu}$ (U235,U238,Pu239)	1	1	1
$\lambda_{capt,fiss,\nu}$ (other fissiles)	1	2	2
λ_{capt} (structurals)	1	1	1
λ_{el} (fissiles and structurals)	1	1	1
λ_{inel} (fissiles and structurals)	1	3	10

Fast Reactor Uncertainty Reduction Requirements to Meet Design Target Accuracies, according to Subgroup 26 (no correlation effects accounted for)

		Energy Range	Current Accuracy (%)	Target Accuracy (%)
U238	σ_{inel}	6.07 ÷ 0.498 MeV	10 ÷ 20	2 ÷ 3
	σ_{capt}	24.8 ÷ 2.04 keV	3 ÷ 9	1.5 ÷ 2
Pu241	σ_{fiss}	1.35MeV ÷ 454 eV	8 ÷ 20	2 ÷ 8
Pu239	σ_{capt}	498 ÷ 2.04 keV	7 ÷ 15	4 ÷ 7
Pu240	σ_{fiss}	1.35 ÷ 0.498 MeV	6	1.5 ÷ 2
	ν	1.35 ÷ 0.498 MeV	4	1 ÷ 3
Pu242	σ_{fiss}	2.23 ÷ 0.498 MeV	19 ÷ 21	3 ÷ 5
Pu238	σ_{fiss}	1.35 ÷ 0.183 MeV	17	3 ÷ 5
Am242m	σ_{fiss}	1.35MeV ÷ 67.4keV	17	3 ÷ 4
Am241	σ_{fiss}	6.07 ÷ 2.23 MeV	12	3
Cm244	σ_{fiss}	1.35 ÷ 0.498 MeV	50	5
Cm245	σ_{fiss}	183 ÷ 67.4 keV	47	7
Fe56	σ_{inel}	2.23 ÷ 0.498 MeV	16 ÷ 25	3 ÷ 6
Na23	σ_{inel}	1.35 ÷ 0.498 MeV	28	4 ÷ 10
Pb206	σ_{inel}	2.23 ÷ 1.35 MeV	14	3
Pb207	σ_{inel}	1.35 ÷ 0.498 MeV	11	3
Si28	σ_{inel}	6.07 ÷ 1.35 MeV	14 ÷ 50	3 ÷ 6
	σ_{capt}	19.6 ÷ 6.07 MeV	53	6

- The Sub26 studies have pointed out that *the present uncertainties on the nuclear data should be significantly reduced, in order to get full benefit from the advanced modeling and simulation initiatives.*

- Only a *parallel effort in advanced simulation and in nuclear data improvement* will enable to provide designers with more general and well validated calculation tools, that would allow to meet design target accuracies

- A further output: new entries in the OECD-NEA High Priority Request List have been proposed, based on uncertainty reduction requirements to meet design target accuracies.

How to meet requirements.

- Some of the most important requirements are difficult to be met using only differential experiments, even if innovative experimental techniques are used.

- The use of integral experiments has been essential in the past to insure enhanced predictions for power fast reactor cores.

- A combined use of scientifically based covariance data and of selected integral experiments can be made using classical statistical adjustment techniques

What is needed

- selection of a set of significant experiments,
- sensitivity analysis of selected configurations including reference design configurations for a wide range of integral parameters
- use of science based covariance data for uncertainty evaluation and target accuracy assessment,
- analysis of experiments using the best methods available, with some redundancy to avoid systematic errors,
- use of calculation/experiment discrepancies in a statistical adjustment

A warning: the credibility of an adjustment is dependent on the credibility of the covariance data and of the experimental uncertainties!

Representativity

- A further use of nuclear data covariance matrix is, in conjunction with sensitivity coefficients, a representativity analysis of proposed or existing experiments.
- The calculation of correlations among the design and experiments allow to determine how representative is the latter of the former, and consequently, to optimize the experiments and to reduce their numbers.

$$\mathbf{r}_{\mathbf{RE}} = \frac{\left(\mathbf{S}_{\mathbf{R}}^{+}\mathbf{D}\mathbf{S}_{\mathbf{E}}\right)}{\left[\left(\mathbf{S}_{\mathbf{R}}^{+}\mathbf{D}\mathbf{S}_{\mathbf{R}}\right)\left(\mathbf{S}_{\mathbf{E}}^{+}\mathbf{D}\mathbf{S}_{\mathbf{E}}\right)\right]^{1/2}}$$

- Formally one can reduce the estimated uncertainty on a design parameter by a quantity that represents the knowledge gained by performing the experiment:

$$\Delta\mathbf{R}_{1}^{2} = \Delta\mathbf{R}_{0}^{2}(1-\mathbf{r}_{\mathbf{RE}}^{2})$$

Statistical Adjustment Method

The method makes use of:

- "a priori" nuclear data covariance information,
- integral experiments analysis to define C/E values
- integral experiment uncertainties
- sensitivity coefficients

If we define: $y_j=(\sigma_j^{adj}-\sigma_j)/\sigma_j$ and $y_{Qi}^{exp}=(Q_i^{exp}-Q_i)/Q_i$, the y_i are given by:

$$y_i = \left(S^T D_Q^{-1} S + D^{-1}\right)^{-1} S^T D_Q^{-1}\, y_{Qi}^{exp}$$

where D_Q is the covariance matrix of the experiments, D the covariance matrix of the cross sections and S is the sensitivity vector. It will also result an adjusted covariance matrix for the nuclear data:

$$\left(D^{adj}\right)^{-1} = D^{-1} + S^T D_Q^{-1}\, S$$

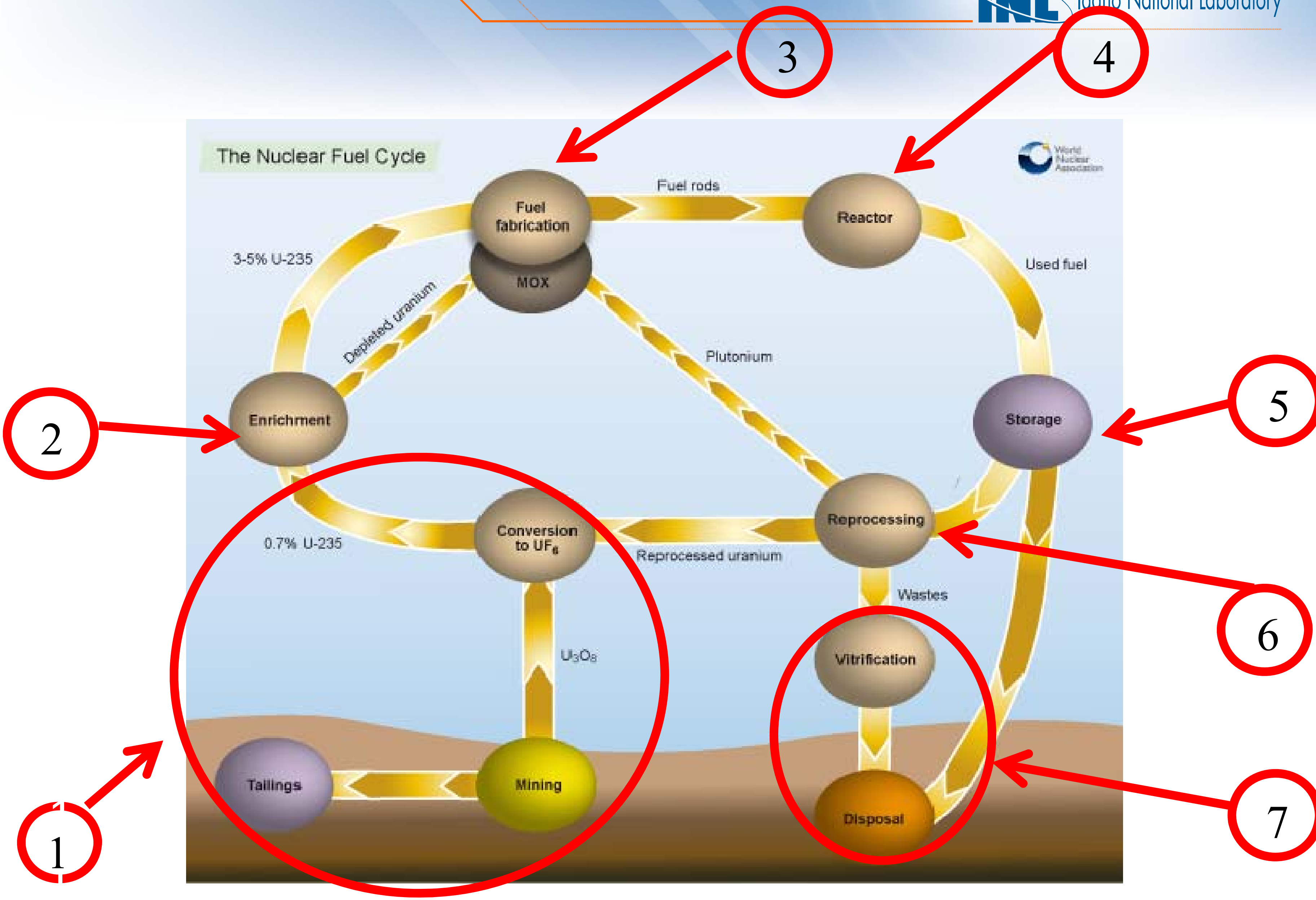
Fuel Cycle

- Many of the central issues associated with nuclear power are tied primarily to the choice of fuel cycle. Resource limitations, non-proliferation, and waste management are primarily fuel cycle issues.

- The fuel cycle provides the mass flow infrastructure that connects the energy resources of uranium and thorium ore through the nuclear power plants to the eventual waste management of the nuclear energy enterprise.

- Natural resources include fuels (uranium and thorium), materials of construction, and renewable resources (such as water for cooling purposes). Wastes may include mill tailings, depleted uranium, spent nuclear fuel (SNF) and high level (radioactive) waste (HLW), other radioactive wastes, releases to the environment (air and water), and nonnuclear wastes.

- Multiple technical facilities are deployed in the fuel cycle. In a simplified fuel cycle schematic, there are ***7 major fuel cycle facilities.***


The Nuclear Fuel Cycle
Fuel fabrication
MOX
Fuel rods
Reactor
Used fuel
3-5% U-235
Depleted uranium
Plutonium
Enrichment
Storage
Reprocessing
Conversion to UF_6
0.7% U-235
Reprocessed uranium
Wastes
U_3O_8
Vitrification
Tailings
Mining
Disposal
1
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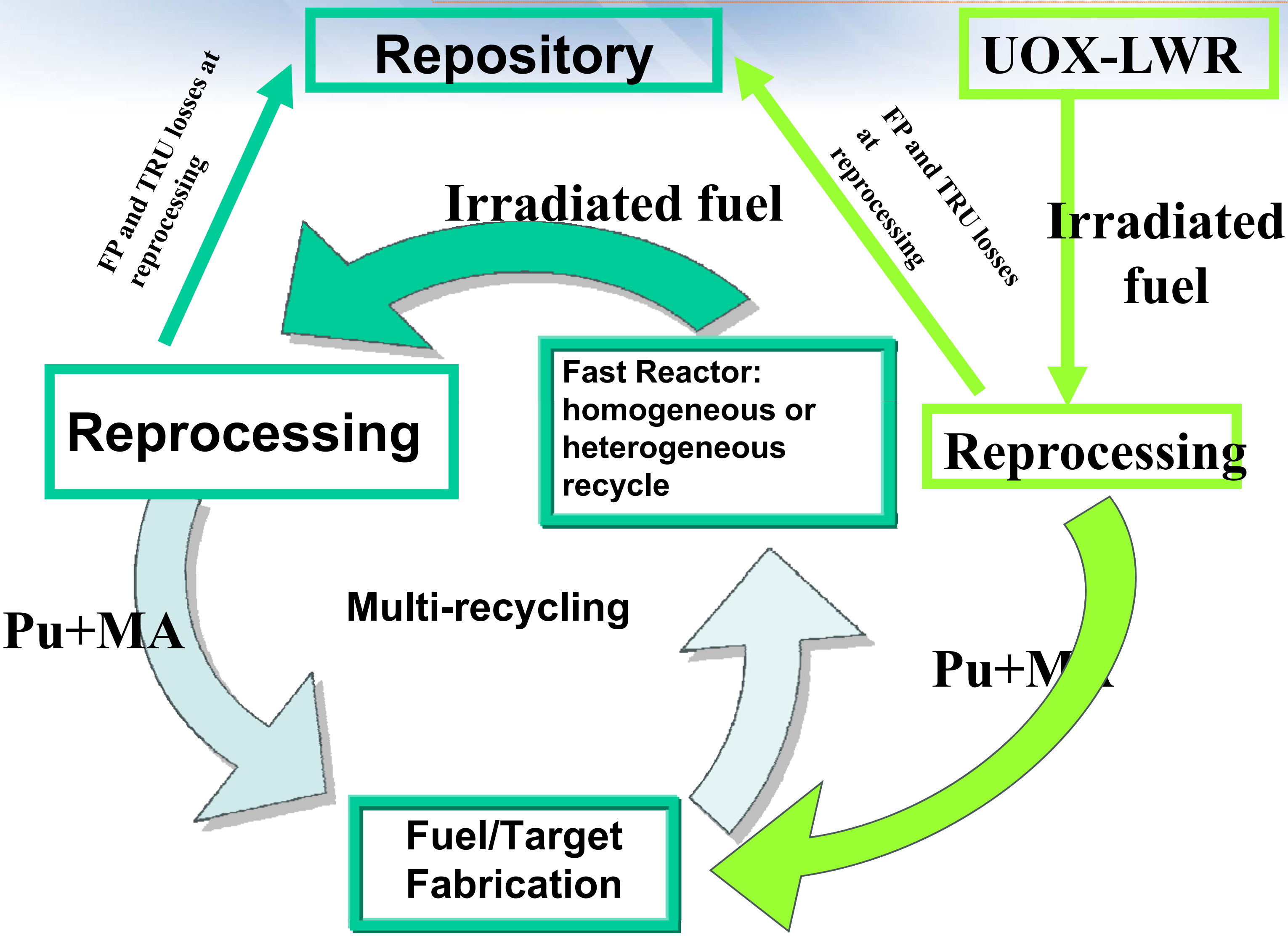


Full-actinide recycle

Simulation needs and challenges for the Fuel Cycle:

1- Nuclei evolution under irradiation and decay outside the reactor:

Bateman equations

- **It is possible to generalize the Bateman equations and account for several operations like reprocessing etc.**

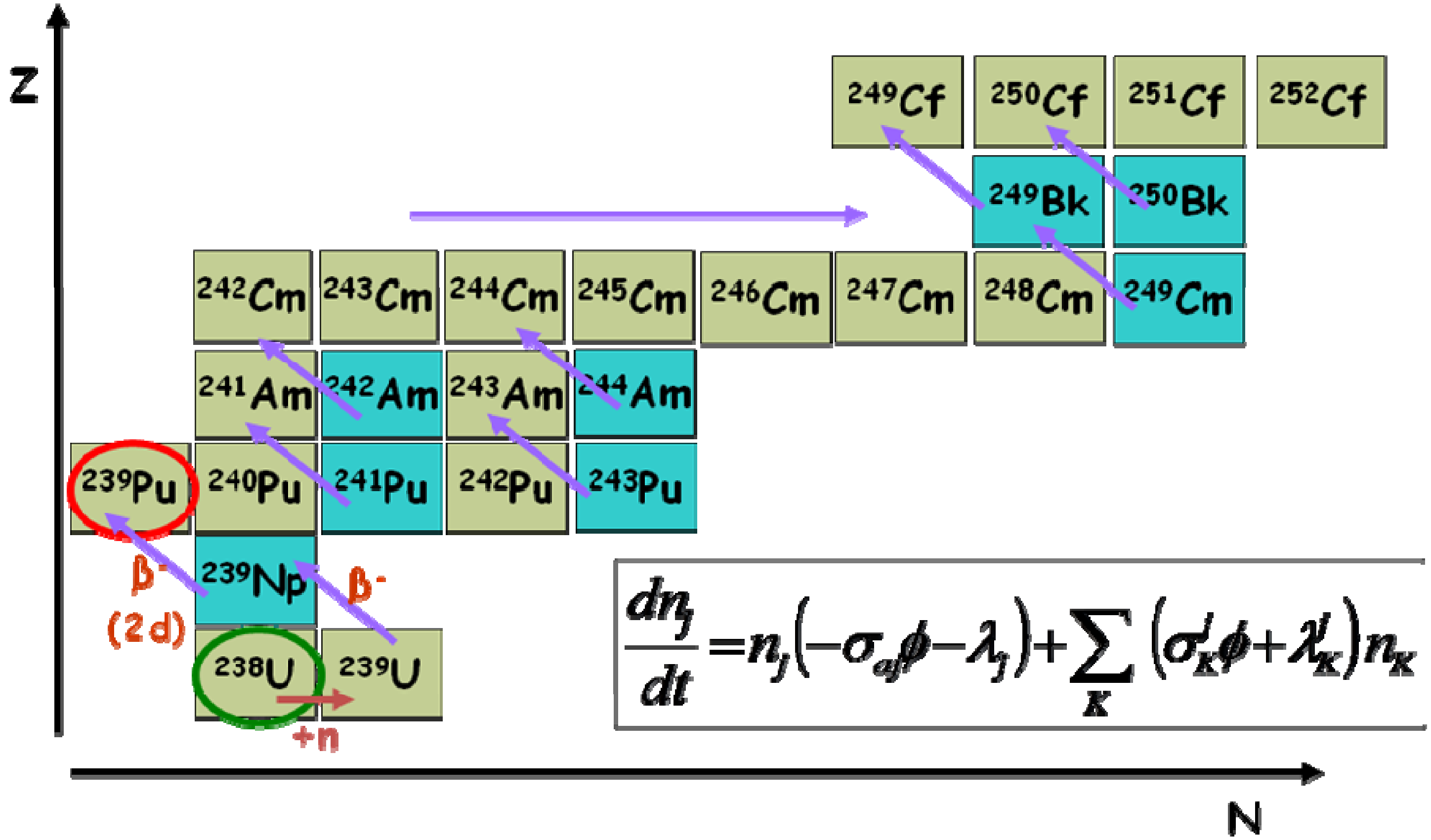
2- Outcome: *nuclei mass inventories, decay heat*, neutron sources, radiotoxicity, doses, radiation protection (e.g. during transport of spent fuel etc), fuel cycle facilities requirements

- **Nuclear data play major role (neutron interaction cross sections, decay data, fission yields etc)**

3- Scenario codes

Nuclei evolution under irradiation

The Uranium nuclei transmutation chain under neutron irradiation and the associated Bateman equations can be represented as follows:

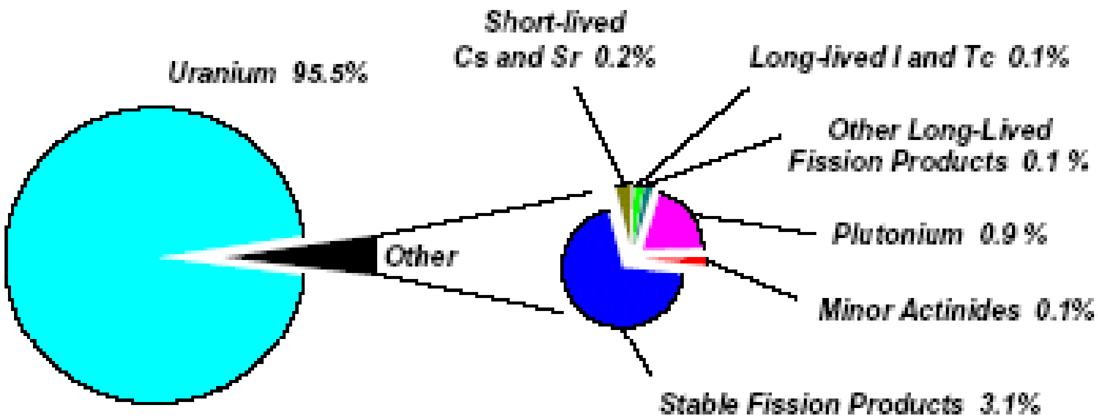


$$\frac{dn_j}{dt} = n_j\left(-\sigma_{aj}\phi - \lambda_j\right) + \sum_K \left(\sigma_K^j\phi + \lambda_K^j\right) n_K$$

where n_j is the nuclide j density, σ_{aj} is the absorption cross section of isotope j, σ_{jK} is the cross section corresponding to the production of isotope K from isotope j, λ_j is the decay constant for isotope j, λ_{jK} is the decay constant for the the decay of isotope j to isotope K and, finally, Φ is the neutron flux.

Application

Composition of Spent Nuclear Fuel (Standard PWR 33GW/t, 10 yr. cooling)



Most of the hazard stems from Pu, MA and some LLFP when released into the environment, and their disposal requires isolation in stable deep geological formations.

A measure of the hazard is provided by the radiotoxicity arising from their radioactive nature.

1 tonne of SNF contains:

955.4 kg U
8,5 kg Pu

Minor Actinides (MAs)
0,5 kg ^{237}Np
0,6 kg Am
0,02 kg Cm

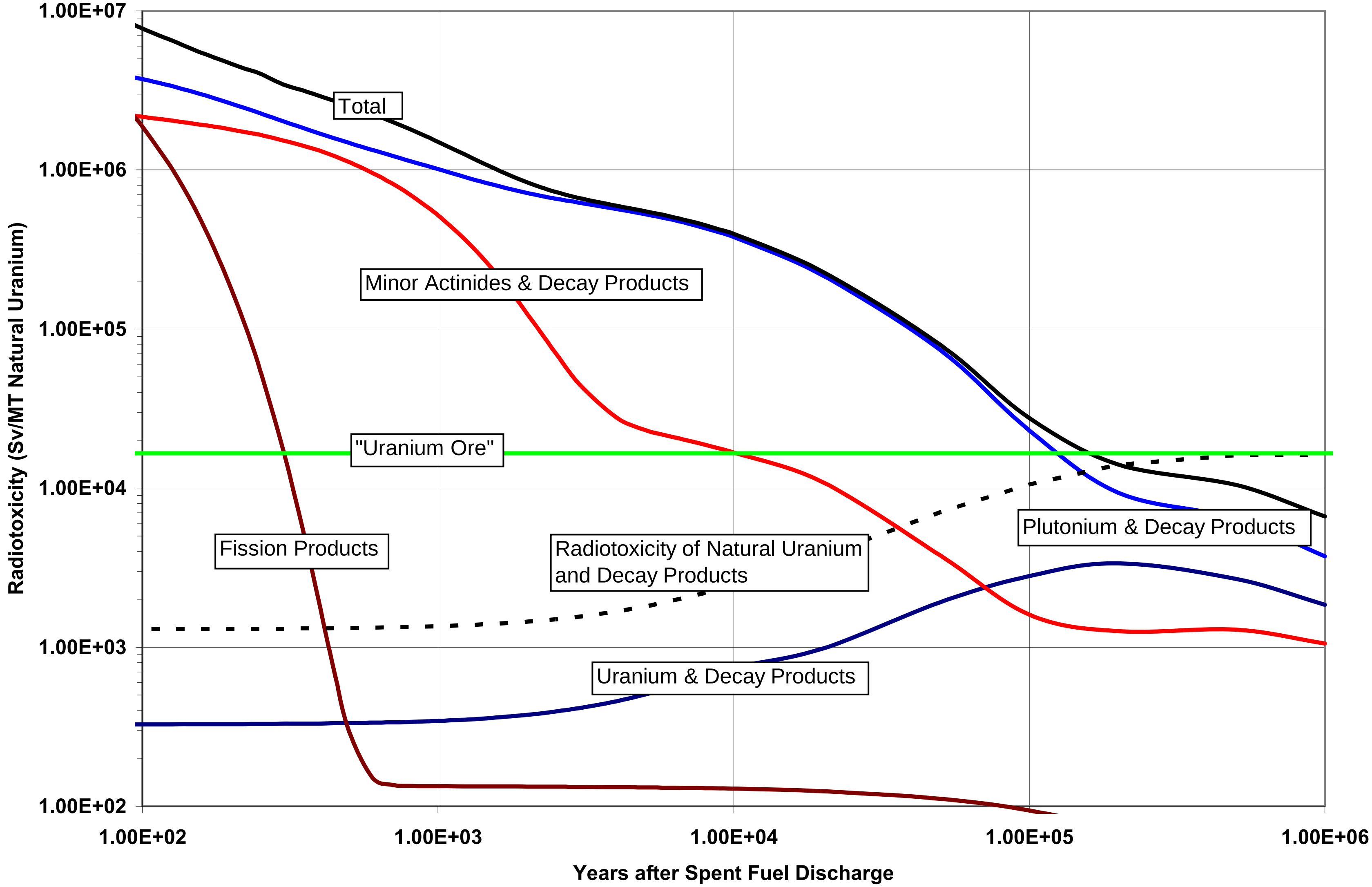
Long-Lived fission Products (LLFPs)
0,2 kg ^{129}I
0,8 kg ^{99}Tc
0,7 kg ^{93}Zr
0,3 kg ^{135}Cs

Short-Lived fission products (SLFPs)
1 kg ^{137}Cs
0,7 kg ^{90}Sr

Stable Isotopes
10,1 kg Lanthanides
21,8 kg other stable

An example of derived quantities: the radiotoxicity

Evolution of the radiotoxic inventory, expressed in sievert per tonne of initial heavy metal (uranium) (Sv/ihmt) of UOX spent fuel unloaded at 60 GW d/t, versus time (years).



Decay Heat: one of the most demanding parameters of the fuel cycle

- This is the delayed heat released from components of nuclear systems after irradiation.
- In reactors this is dominated by the fuel assembly components (includes heavy elements, fission products, and activation products)
- Results from beta and alpha decay, internal transitions and spontaneous fission of nuclides present.
- Includes:
  - photons (x-rays and gamma),
  - leptons (electrons and positrons) and
  - baryons (alpha particles, neutrons, nucleus recoil)

Decay Heat: Some examples of the most important problems.

Accidental situation:

- Determination of cooling needs in the reactor after shutdown.
- Evaluation of radiation doses:
  - inside the reactor to study the accessibility of staff (a long term problem) and maintenance of electrical and mechanical (short-term problem) equipment.
  - in the surrounding of the plant in case of leakage of radiation.

In normal operation:

- Safety of gamma thermometry instrumentation which measures, in stable and transient regime, the local power of the reactor.
- In the immediate surroundings of the core or of the spent fuel casks.

In fuel cycle (out of pile):

- Determination of cooling needs in spent fuel pools
- Doses at the different installations of the fuel cycle (e.g. neutron sources at fuel fabrication)
- Decay heat in a repository (this was the dimensioning parameter for the Yucca Mountain repository)

Decay Heat Validation

How accurately can decay heat be estimated?

- Short term (seconds to days)
  - Reactors
  - Many nuclides important
- Mid term (years to decades)
  - Storage, transport, chemical processing
  - Small number of nuclides important
- Long term (centuries to millions of years)
  - Geological disposal
  - Few nuclides important

Besides actinides, fission products play an essential role

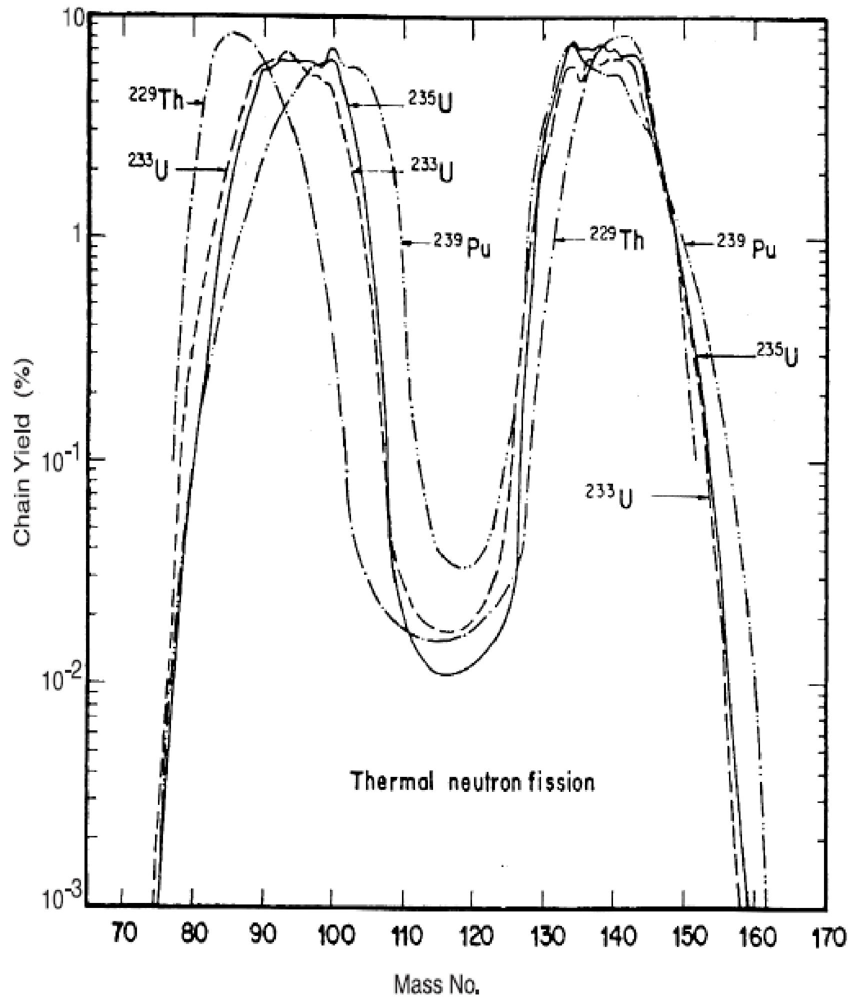


Fig. 8. Mass distribution curves in the thermal-neutron induced fission of ^{229}Th, ^{233}U, ^{235}U, and ^{239}Pu

When the actinide and fission-product inventories have been calculated for the specified conditions of reactor operation and subsequent cooling period, the decay heat can be derived by summing the products of the nuclear activities in terms of the mean alpha, beta and gamma energy releases per disintegration of that nuclide:

Decay Heat Components

$$H_\alpha(t) = \sum_{i=1}^{M} \lambda_i^T N_i(t) E_\alpha^i$$

$$H_\beta(t) = \sum_{i=1}^{M} \lambda_i^T N_i(t) E_\beta^i$$

$$H_\gamma(t) = \sum_{i=1}^{M} \lambda_i^T N_i(t) E_\gamma^i$$

where E_α^i , E_β^i and E_γ^i are the mean alpha, beta and gamma energy releases respectively per disintegration of nuclide i; λ_i^T is the total decay constant of nuclide i, and $H_\alpha(t)$, $H_\beta(t)$ and $H_\gamma(t)$ are the total alpha, beta and gamma decay heat respectively at time t after reactor shutdown.

Differences in decay heat contributions

UOX

- 5 years

Ba137m	22.8% (7.1E4)
Y90	18.5%
Ru106	13.3%
Cs134	8.0%

- 10 years

Ba137m	28.9% (6.3E4)
Y90	23.4%
Am241	15.1%

MOX

- 5 years

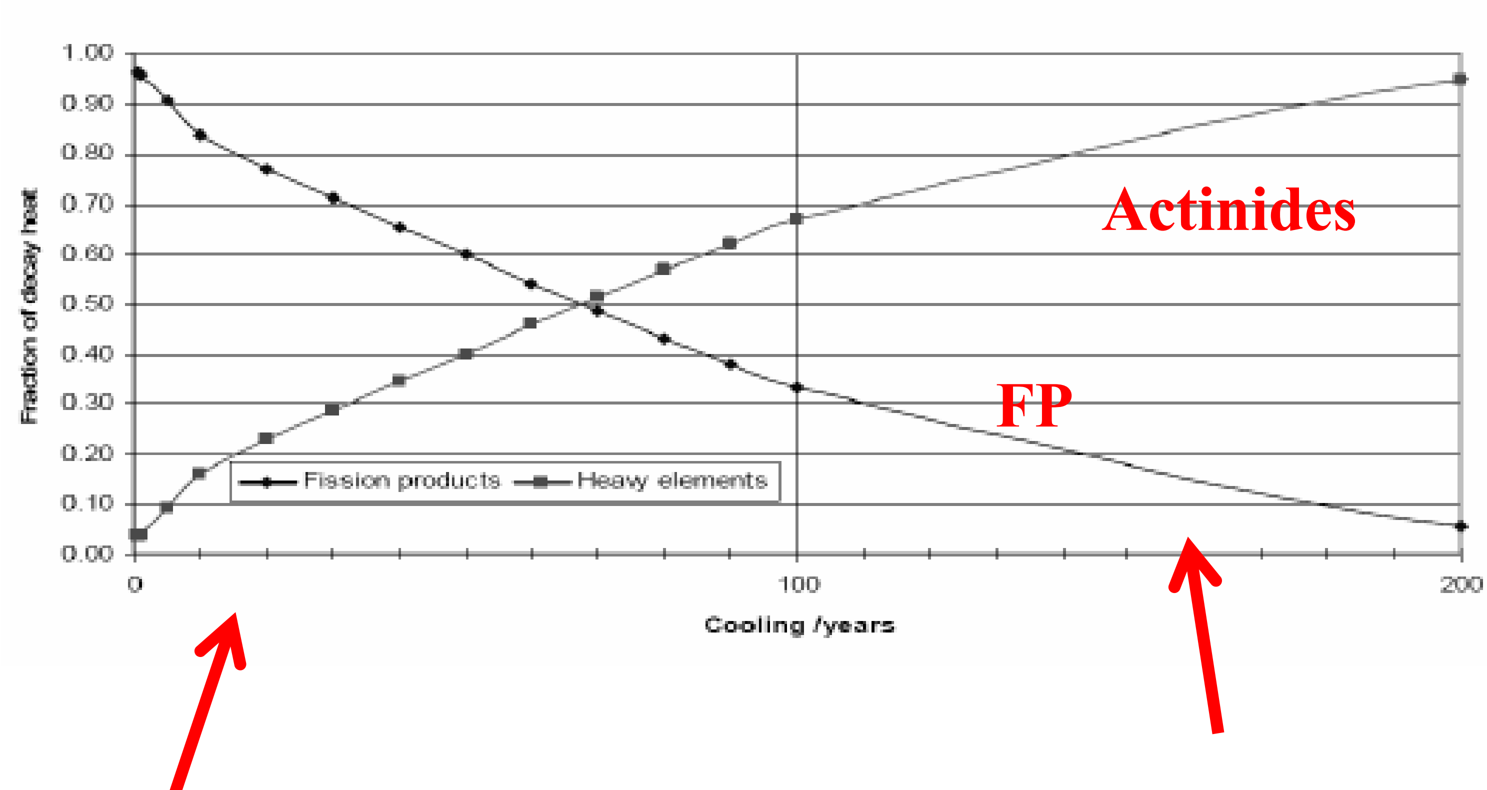
Pu238	29.1%
Cm244	27.2%
Ba137m	7.9% (7.0E4)

- 10 years

Pu238	34.3%
Cm244	27.5%
Am241	12.7%
Ba137M	8.7% (6.2E4)

Relative role of FPs and Actinides for a standard LWR

- Decay heat from fission products and heavy elements (Z>80)



Short cooling times

Long cooling times

Uncertainties can be much higher for innovative fuel cycle with full MA recycle. E.g. in the case of a fast reactor (SFR) loaded with a TRU fuel with MA/Pu ratio ~0.1, the decay heat is soon dominated by higher Pu isotope and MA contributions.

Example: relative contribution by isotope (%) for the SFR reactor on the decay heat in the repository
100 years after disposal

A challenge for simulation codes!

Isotope	Decay Heat
Np237	-
Pu238	46.51
Pu239	1.37
Pu240	6.82
Pu242	0.03
Am241	26.61
Am242m	0.17
Am243	0.65
Cm242	12.91
Cm243	0.10
Cm244	4.59
Cm245	0.10
Cm246	0.10
Cm248	-
Total	100.00

The modelling of the fuel cycle (i.e. evaluation of nuclei densities and e.g. decay heat) allows to evaluate the impact of full actinide recycle strategies on fuel cycle parameters:

Reactor type	PWR		FR					ADS
Fuel type / Parameter	MOX (Pu only, reference)	Full TRU recycle	Pu only	Homog. TRU recycle, CR=1 and MA/Pu~0.1	Homog. TRU recycle, CR=0.5 and MA/Pu~0.1	Homog.TRU recycle, CR=0.5 and MA/Pu~1	MA targets (Heterog. Recycle, 10-20% MA in the targets)	MA-dominated fuel CR=0 and MA/Pu~1
Decay heat	1	x3	x0.5	x2.5	x12	x38	x40	x100
Neutron source	1	x8000	~1	x150	x1000	x4000	x5000	x20000

Different strategies of full actinide recycle

In conclusion:

- New innovative systems (reactors and fuel cycles) will likely present specific features that are very sensitive to nuclear data uncertainties. This is probably also the case of innovative thermal reactors (e.g. VHTR)
- In preliminary phases of conceptual design scoping, larger uncertainties can probably be tolerated
- However, in further consolidated design phases, low uncertainties and sound correlation data are required for feasibility, safety, and economic reasons
- There are challenging issues that can only be coped with the use of robust, science-based covariance data and high accuracy integral experiments