European Joint Programme on Radioactive Waste Management

Spent Fuel Characterization and evolution until disposal

Task 2: Fuel properties characterisation and related uncertainty analysis

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Participants:

CIEMAT, CPST, CTU (SURAO), ENRESA, ENUSA, JRC, JSI, KIT, LEI, NAGRA, PEL, PSI, SCK•CEN, SKB, SSTC NRS, TUS, VTT, UU

SNF - Gent, 15/11/2019

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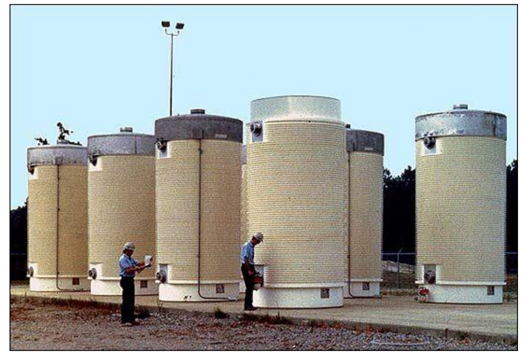
SFC: Spent Fuel Characterization and Evolution Until Disposal

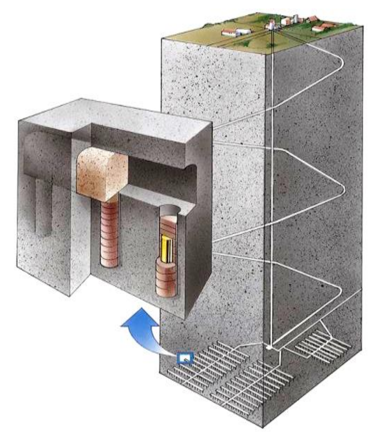
- Task 1: S/T coordination, state-of-the-art and training material
- **Task 2: Fuel properties characterisation and related uncertainty analysis**
- *Task 3: Behaviour of nuclear fuel and cladding after discharge*
- Task 4: Accident scenario and consequence analysis
- Task 5: Civil Society interaction

Spent Nuclear Fuel (SNF) intermediate storage or final disposal

A **safe**, **secure**, **ecological** and **economic** transport**,** storage and final disposal requires that **SNF** is **characterised** for the main source terms of interest:



- Decay heat : H
- Neutron emission : S_n
- γ-ray emission : S_γ
- Reactivity : R
- Fissile material : Nuclear Safeguards
  i.e. ^{235}U, ^{239}Pu
- Specific nuclides : Long term safety
  e.g. ^{14}C, ^{79}Se, ^{94}Nb, ^{99}Tc, ^{129}I, ^{226}Ra



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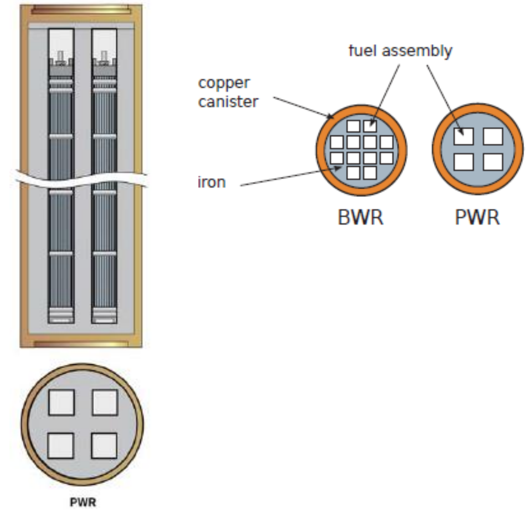
Characterisation of SNF

Main **source terms** of interest:



- Decay heat : H
- Neutron emission : S_n
- γ-ray emission : S_γ
- Reactivity : ^{235}U, ^{239}Pu, Fission Products (BUC)
- Fissile material : ^{235}U, ^{239}Pu
- Long-term safety : e.g. ^{14}C, ^{79}Se, ^{94}Nb, ^{99}Tc, ^{129}I, ^{226}Ra

difficult to be measured directly, in particular during industrial operation

- Decay heat by calorimetry at CLAB: accurate but long measurement times
- Criticality safety analysis: calculations required to account for burn-up credit (MA, FP)

⇒ Determined/estimated by **theoretical calculations** using a burnup code

Neutron transport + depletion/creation code (see L. Fiorito)

e.g. decay heat of SNF

Decay heat rate: $H(t) = \sum_j H_j(t)$

- $H_j(t)$: contribution of radionuclide j

  $H_j(t) = h_j\, N_j(t)$

  – h_j : specific decay heat rate of nuclide j

  – $N_j(t)$: number of nuclei of nuclide j at time t

- $N_j(t = t_0)$ known at $t = t_0$ (after irradiation)

  – $N_j(t > t_0)$ can be predicted accurately

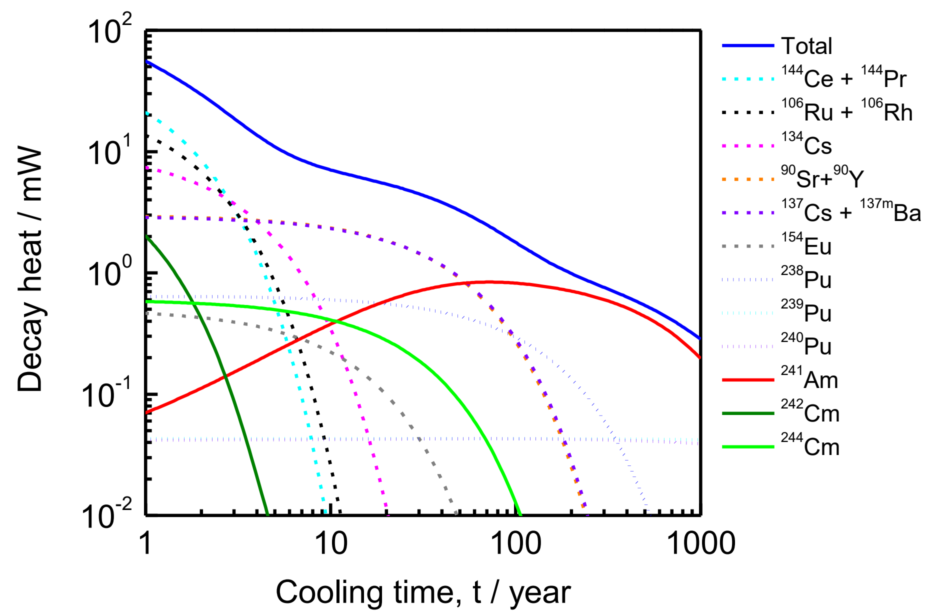
  $N_j(t) = N_j(t_0)\, e^{-\lambda_j (t-t_0)}$

PWR UO_2 pellet

$^{235}U/U = 4.8$ %

m(U) = 4.4 g

burnup = 44 GWd / tU



$\Rightarrow$ accurate nuclide vector $N_j(t_0)$ at t_0 is required from burn-up codes

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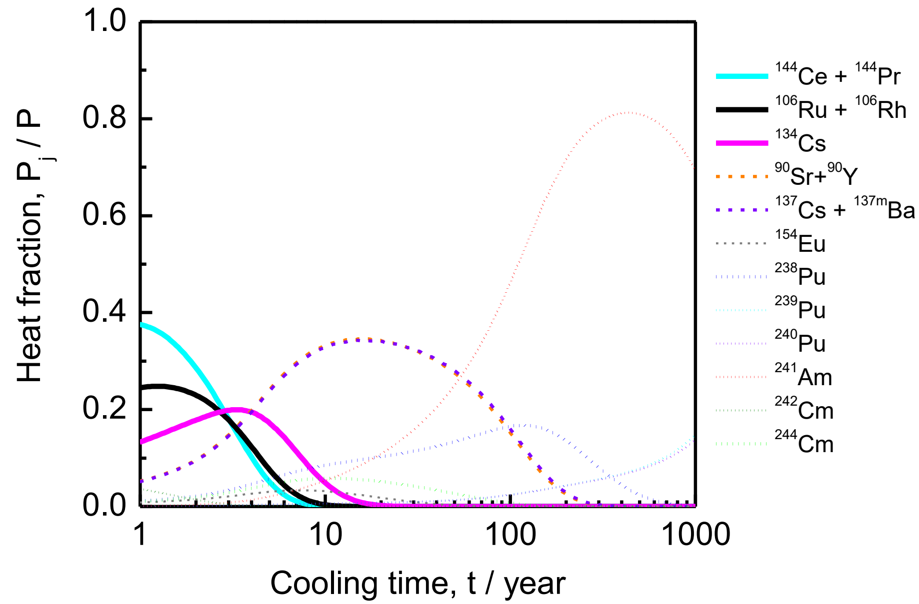

$\Rightarrow$ **accurate nuclide vector $N_j(t_0)$** at t_0 is required from **burn-up codes**

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Task 2: Fuel properties characterisation and related uncertainty analysis

- **Subtask 2.1 (Theory)** co-ordinator PSI
  Theoretical study of SNF source terms (assess and improve performance)

- **Subtask 2.2 (NDA)** co-ordinator SCK•CEN
  Develop, improve and demonstrate NDA methods/systems for SNF characterisation

- **Subtask 2.3 (Cladding)** co-ordinator KIT
  Calculate and determine experimentally the inventory of activation and fission products in cladding material

- **Subtask 2.4 (Final procedures)** co-ordinator KIT (PEL)
  Define and verify procedures to determine the source terms of SNF with realistic confidence limits

Subtask 2.1: Theoretical study of SNF source term

2.1 Theory

Compare codes
- State-of-the art codes
- Selection of SNF assemblies
  (well-known history)

Sensitivity analysis
- Nuclear data
- Fuel properties
- Operation history

Outcome
- Status codes
- Nuclear data needs
- Crucial SNF history parameters

Subtask 2.1 – Code comparison

• NAGRA	:	SCALE
• JSI	:	SCALE, DRAGON, (MCNP/FISPACT)
• PSI	:	CASMO
• VTT	:	SERPENT
• JRC Geel	:	SERPENT
• CIEMAT	:	MCNP/ACAB
• SCK•CEN	:	MCNP/ALEPH-2

Subtask 2.1 – SNF assemblies for code comparison

- 3 **PWR** assemblies
  - SF95-5 from the Takahama-3 reactor
    (UO_2 assembly, 30 MWd/kgU, 2 cycles, 4.1 %). Details in **SFCOMPO**
  - BM1 from the ARIANE campaign
    (MOX assembly, 45-47 MWd/kgU, 5 cycles, 4.2 % fissile). Details in **ARIANE** public report
  - GU1 from the ARIANE campaign
    (UO_2 assembly, 60 MWd/kgU, 4 cycles, 3.5 %). Details in **ARIANE** public report

- 2 **BWR** assemblies
  - KLU1 from the MALIBU campaign
    (UO_2 assembly, 60 MWd/kgU, 7 cycles, 3.9 %). Details **publicly available** 01/2020
  - Gundremmingen-7 sample
    (UO_2, 25 MWd/kgU, 4 cycles, 2.5 %). Details in **SFCOMPO**

- **Subtask 2.2**
  - ENRESA (BWR)
  - **SKB-50** campaign + blind test SKB: 2 PWR assemblies
  - **Reference pellet (REGAL)** (PWR UO_2 , 4.25 %, 50 MWd / kgU)

Subtask 2.1: Sensitivity and covariance analysis

- Nuclear data
  - Cross sections
  - Fission yields
  - Decay data (emission probabilities, recoverable energy per reaction/decay)
- Fuel properties: geometry and material data
  - Initial enrichment
  - …
- Operational history
  - Burn-up
  - Cooling time
  - Influence of neighbouring assemblies
  - Albedo boundary conditions
  - …
- Computational
  - Method (stochastic/deterministic)
  - Model (2D/3D, boundary conditions, etc.)
  - Numerical approximations (depletion time steps, depletion zones, etc.)

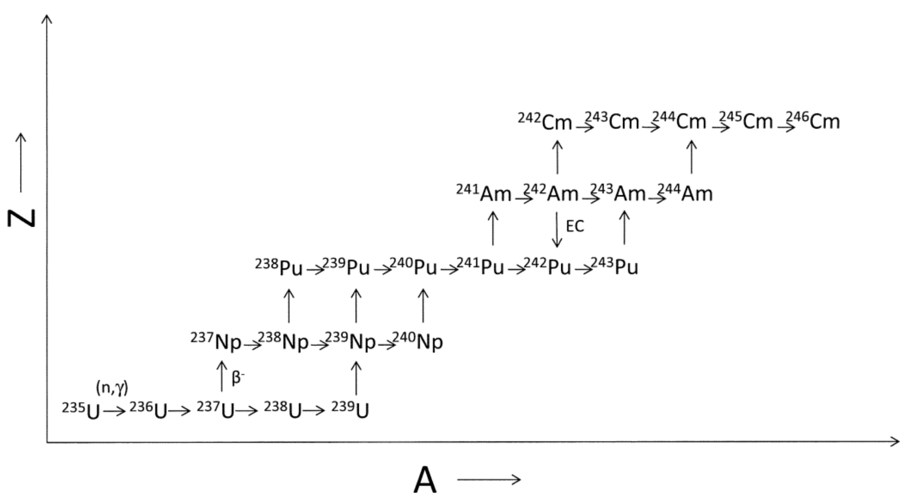
$\Rightarrow$ Recommendations for improvement and realistic uncertainty limits

Sensitivity analysis: e.g. $^{242,\,244}$Cm production in MOX

- MOX, Borella et al. MC 2017, Jeju, Korea
  - PWR: 17 x 17
  - Single pin model

$$S_{N\sigma} = \frac{\partial N / N}{\partial \sigma / \sigma}$$

- Production : σ(n,γ)
  - ^{242}Cm : 2.5 % ^{241}Am(n,γ), ^{242m}Am(n,γ)
  - ^{244}Cm : 10 % ^{242}Pu(n,γ), ^{243}Am(n,γ)



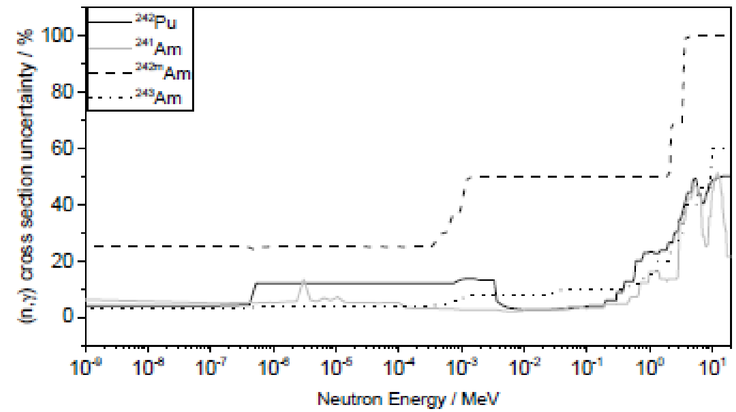


Fig. 4. Relative uncertainties of some of the considered radiative capture cross sections.

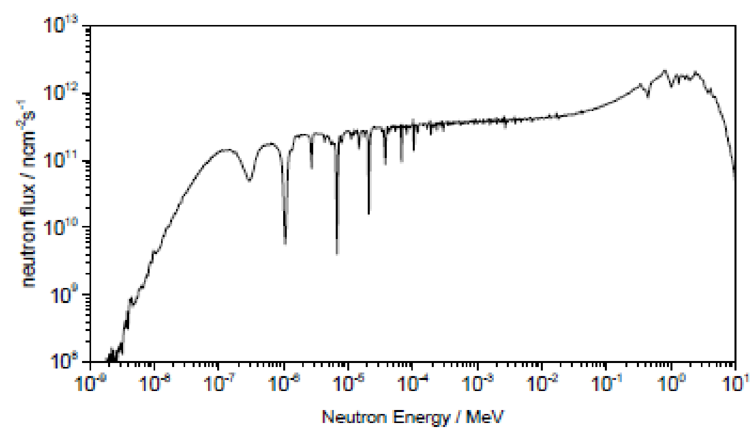
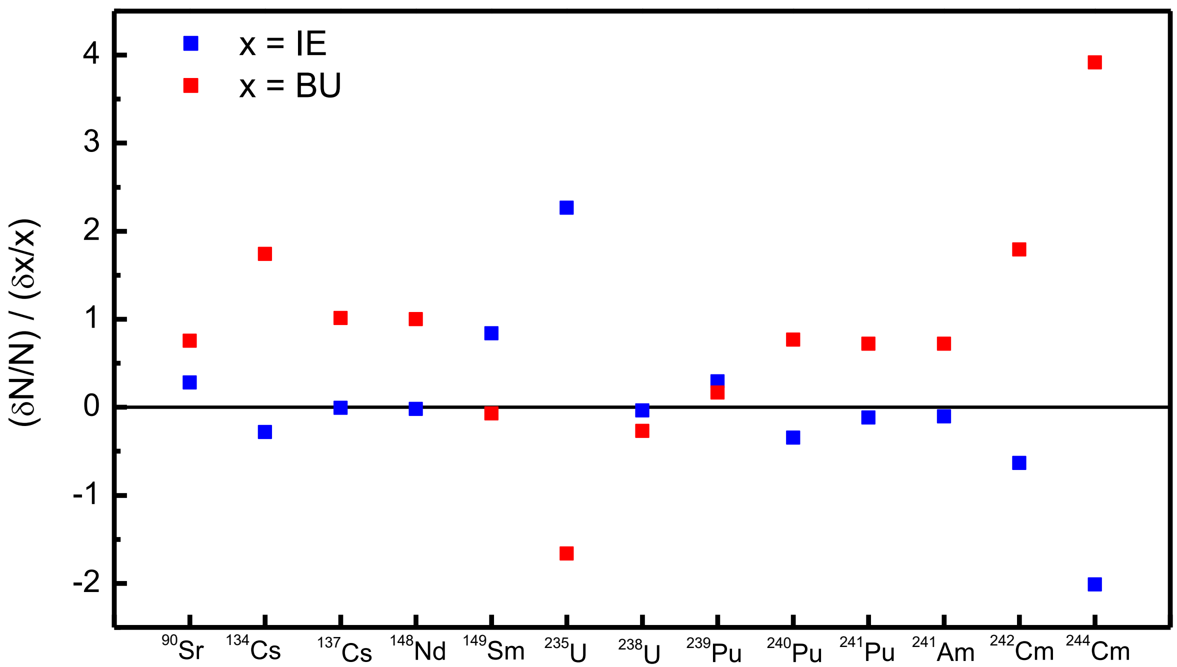


Fig. 5. Neutron flux spectrum in the studied model.

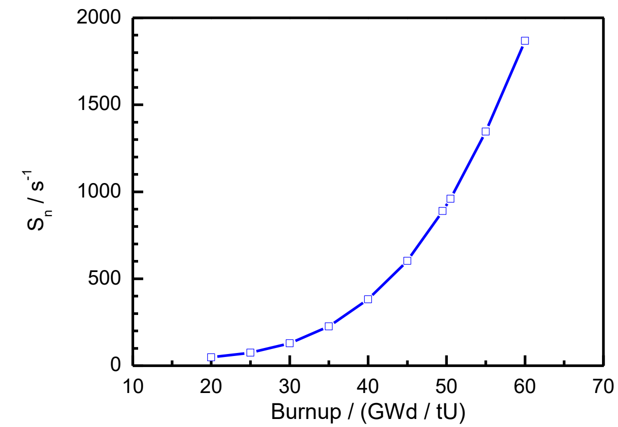
Sensitivity analysis: initial enrichment and burnup



PWR UO_2

$^{235}U/U = 4.8$ %

burnup = 50 MWd/kgU



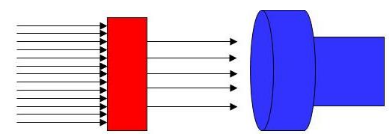
Subtask 2.2: NDA methods/systems for SNF characterisation

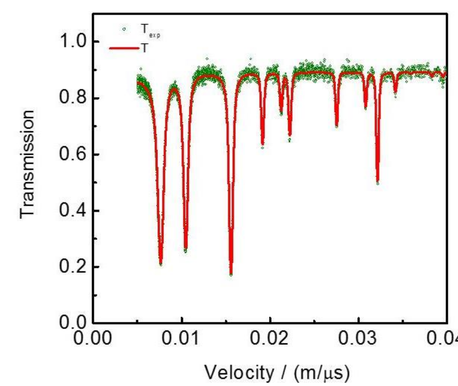
- **Innovative NDA** methods to characterise **small samples**
  - Validate codes (alternative to radiochemical analysis)
  - Production of a reference pellet
- **NDA** methods to characterise **fuel assemblies** (SKB-50)
  - Validate codes
  - Improve theoretical predictions during industrial routine operation
- Study new detectors
  - CLYC, *CVD*
- *Radiochemical analysis of BWR samples*

$\Rightarrow$ input to 2.1 and 2.4

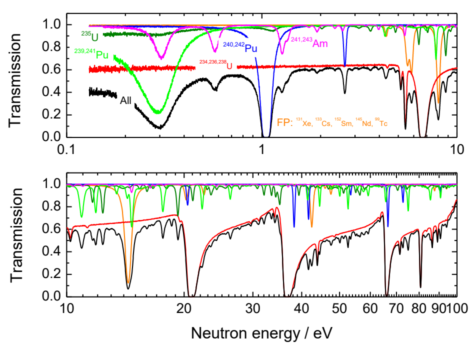
NDA methods to characterise small samples

- **Neutron emission rate** of a SNF pellet
  - Non-destructive method to determine ^{244}Cm content
  - Measurements in conventional controlled area conditions
  - First measurements at SCK•CEN Mol using an existing neutron counter developed and calibrated at JRC finalised
  - Improved/optimised neutron counter under development at JRC Ispra (optimised to apply Hage's point model)





- **Nuclide vector** of SNF pellet by **NRTA**
  - Non-destructive; no chemical analysis
  - Absolute measurement
  - Measurements at GELINA facility of JRC Geel



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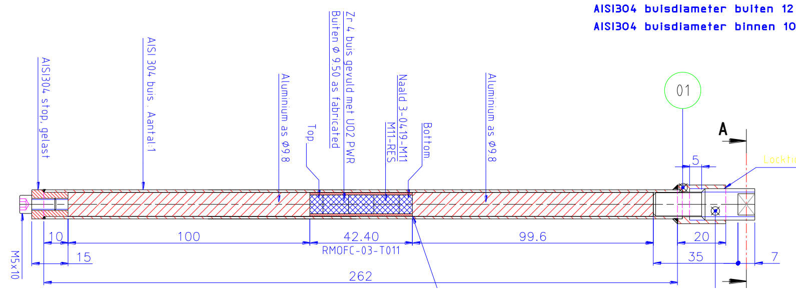
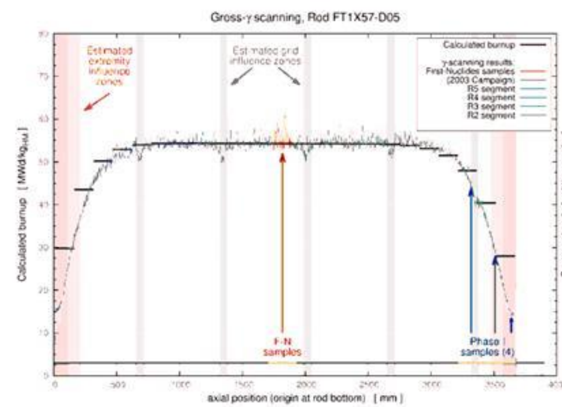
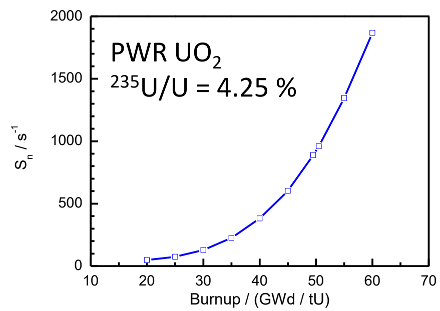
Reference spent nuclear fuel pellet

- Material: **REGAL project**
  - Well-known production and irradiation history
  - γ-ray scan of fuel pin
  - Neighbouring pellets: analysed by radio-chemistry





- NDA measurements on **reference SNF pellet**
  - γ-ray spectroscopy
  - **Neutron emission rate**
  - **NRTA: nuclide inventory (GELINA)**



PWR UO_2

$^{235}U/U = 4.25$ %

burnup = 50 GWd / tU

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Fuel assemblies: calorimeter at CLAB

Improve and assess performance of calorimeter

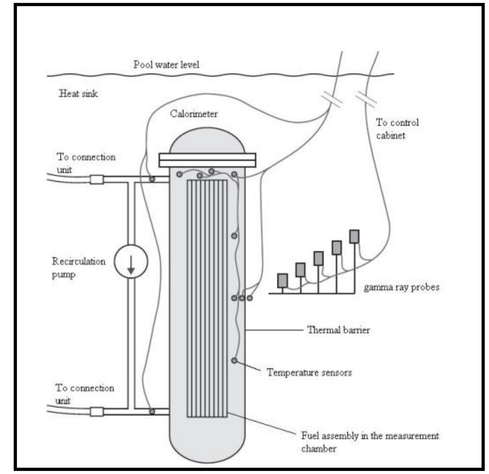
- Performance assessment
  - Understand measurement method (analytic model)
  - Dedicated experiments to apply ANOVA and MLE

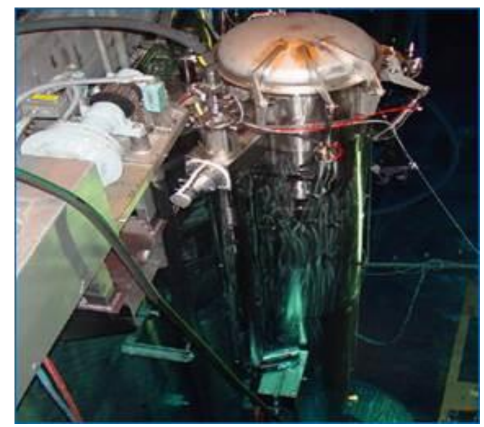
    ANOVA : Analysis Of Variance
    MLE : Maximum Likelihood Estimate (Bayesian approach)

- Calibration
  - Difference electrical heater ⇔ nuclear fuel
  - Heat losses due to escaping γ-rays

⇒ **Reference instrument** for code validation
(SKB-50 input for 2.1 and 2.4)

⇒ Objective: decay heat ~ 2%





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Fuel assemblies: DDSI and DDA (LANL development, NGSI)

Improve/extend data analysis of DDA and DDSI (part of the SKB-50 project)

- Differential Die Away Self-Interrogation (DDSI, passive) *A.C. Trahan, LA-UR_16_20026*

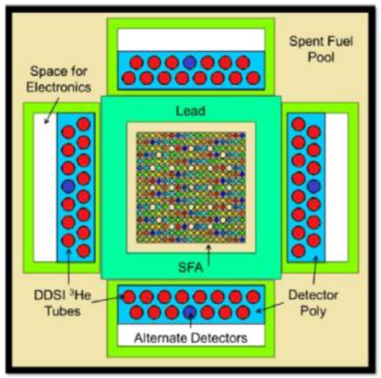
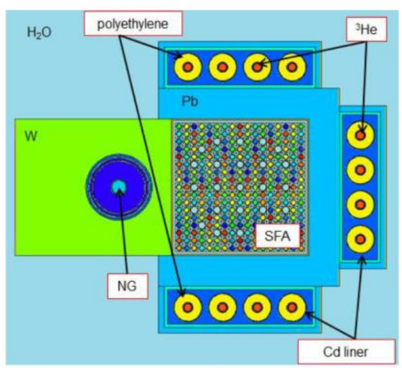


Fig. 1. Cross-sectional view of the simulated DDSI instrument.

- Differential Die Away (DDA, active) *V. Henzl, LANL-UR-123025*



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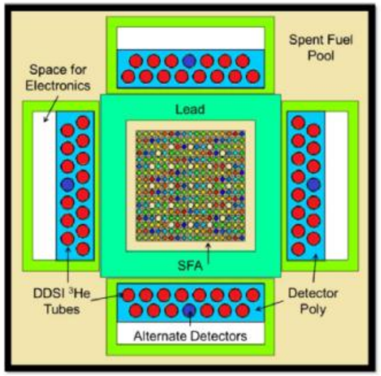
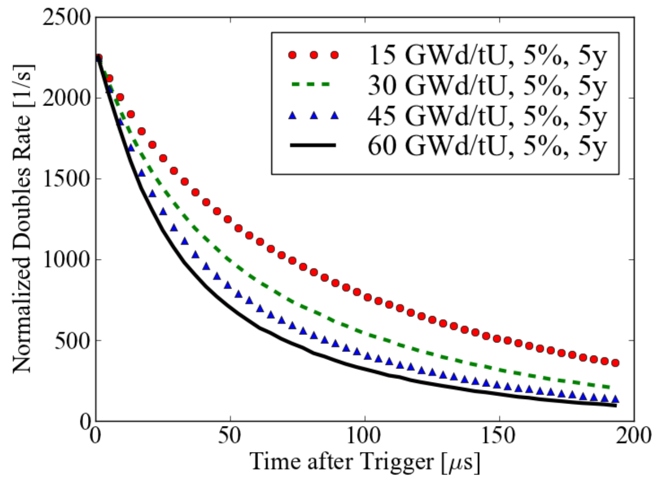
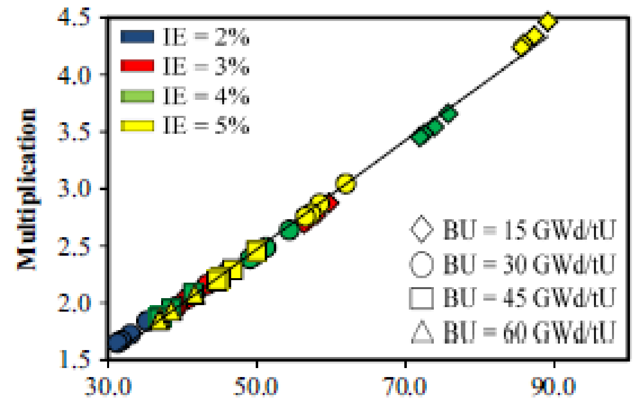
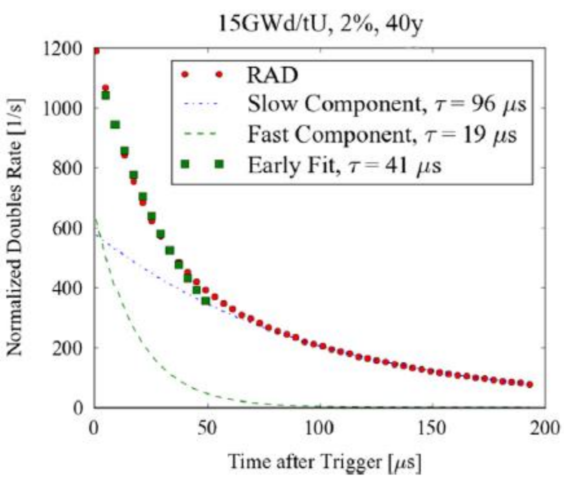






Die-away time/ μs



Optimise the use of data:

- Analyse the full Rossi-Alpha distribution
- Determine source strength: ^{244}Cm

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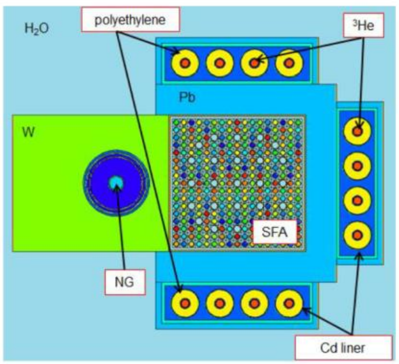

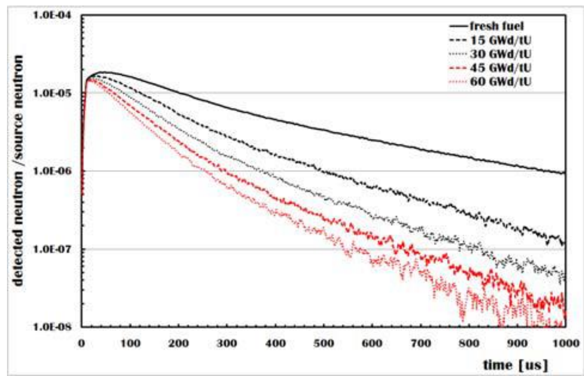

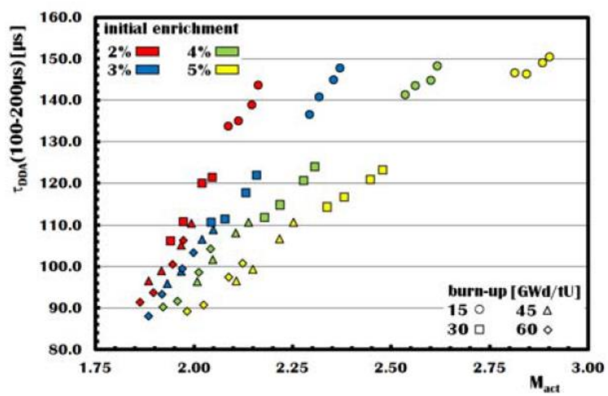

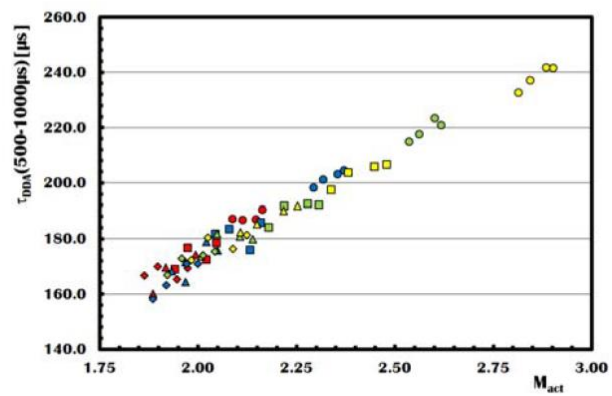

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Subtask 2.3 – Inventory of activation products and FP in cladding

- Preparation of three irradiated samples for further analysis (2 types)
  - Cladded pellets of UO_2 (50 GWd/t_{HM}) and MOX (38 GWd/t_{HM}) irradiated in a PWR
  - Plenum cladding from a UO_2 (50 GWd/tHM) rod segment irradiated in a PWR

- Experimental determination of nuclide inventory: radiochemical analysis
  - FP + ^{14}C and ^{36}Cl

- Validate and improve theoretical calculations of nuclide inventory

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Subtask 2.4: Final objective

2.1 Theory

2.2 NDA

2.3 Cladding

2.4 Define procedures to determine SNF source terms

- Compare best practice industrial and most advanced code
- Define procedure to characterise SNF in industrial conditions based on:
  - Best practice industrial code
  - Realistic NDA measurements (time, industrial environment)

Thank you for your attention