eurad 2
European Partnership
on Radioactive Waste Management

Topical Session 3- SAREC and CSFD
Deep Geological Disposal of Spent Nuclear Fuel: Radionuclide Release and Criticality Safety Considerations

Lena Z Evins (SKB), WP8 Sarec &
Madalina Wittel (Nagra), WP17 CSFD

EURAD-2 Annual Event N°2

8 – 10 September 2026, Dublin, Ireland

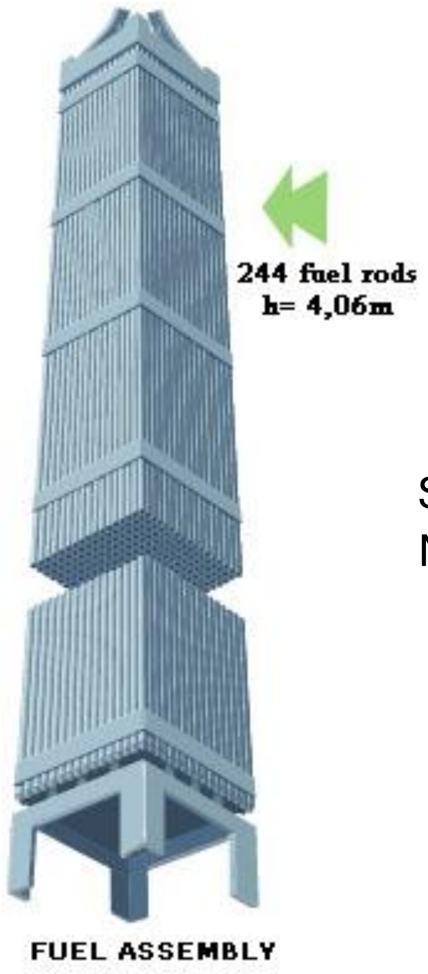
Co-funded by the European Union under Grant Agreement n° 101166718

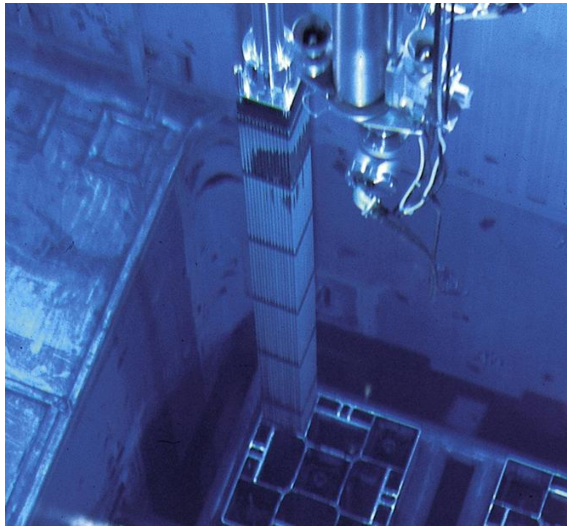
AGENDA

Time	What	Who
14:00-14:15	Introduction	Lena Z Evins (SKB, SE), Madalina Wittel (Nagra, CH)
14:15-14:45	Results from both WP8 + WP17	Lena Z Evins (SKB, SE), Madalina Wittel (Nagra, CH)
14:45-15:00	Highlight on joint issues	Thierry Mennecart (SCK CEN, BE), WP8 participant
15:00-15:15	End Users perspective	Miguel Cuñado (Enresa, ES)
15:15- 15:30	Scientific discussion with audience involvement	All (Led by Lena & Madalina)

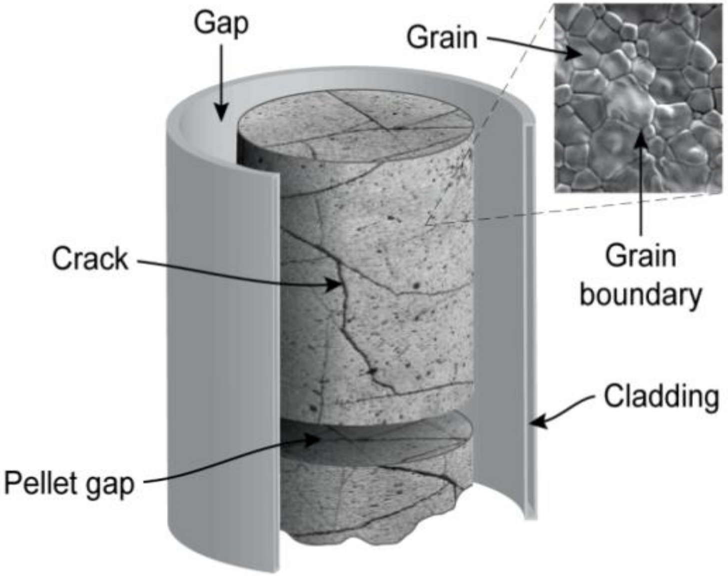
Rapporteur: Olivia Roth (SSM)

INTRODUCTION (I) - NUCLEAR FUEL OVERVIEW



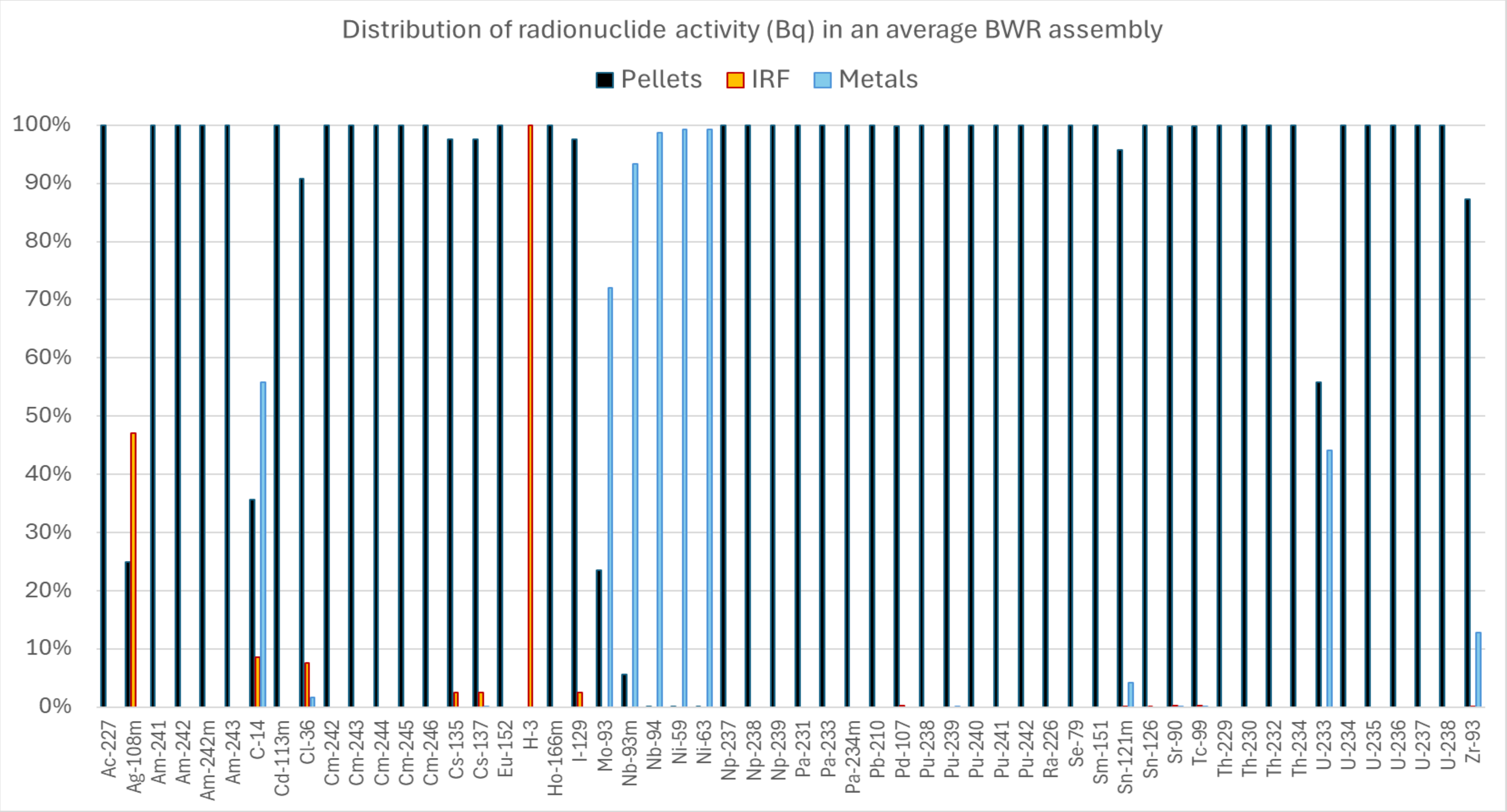


Standard oxide fuel: UO_2
MOX: $(U, Pu)O_2$



Used UO_2 fuel

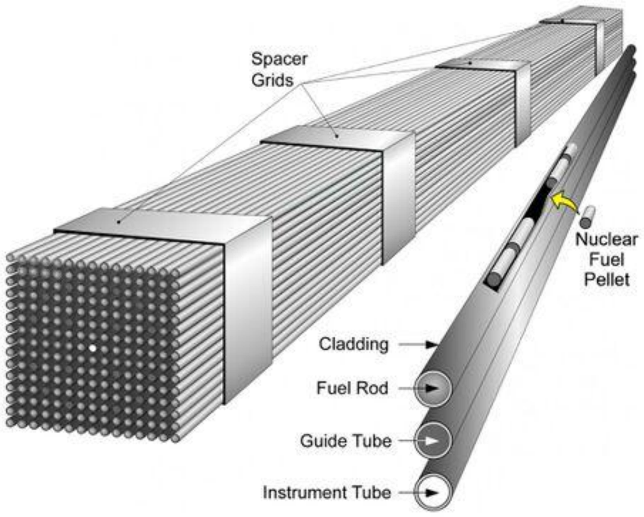
INTRODUCTION (II) - IRRADIATION & SPENT NUCLEAR FUEL



INTRODUCTION (III) – RN INVENTORIES

- The isotopic composition of irradiated nuclear fuel comprises a long list of radionuclides
- Depending on the application, there are different approaches to identifying the list of relevant radionuclides and to deriving the amount present in spent fuel.
- For DGR safety assessments, we rely on calculated inventories: the fuel irradiation can be simulated to derive the final isotopic composition with dedicated computer codes such as:
    - MCNP or SCALE
    - CMS
    - MONK
    - etc.
- The calculations are typically validated against experimental data
    - The SF-COMPO database
    - Ongoing experimental programmes
- Synergy between WP8 & WP17:

The "source term": WP-8 is concerned with the amount of radionuclides released, while WP-17 is concerned with what remains.



INTRODUCTION (V) – CRITICALITY IN A FINAL REPOSITORY

- Irradiated nuclear fuel still contains certain amounts of fissile material (nuclides)
  - The half-lives of these nuclides cover a very wide range: e.g. 14 years (Pu-241), 704 million years (U-235), etc.
- Under very specific circumstances, this could potentially lead to new fission chain reactions in the deep geological repository (DGR) → nuclear criticality.
- **Criticality safety of the DGR is a safety requirement in all national programmes that must dispose of high-level waste, such as spent fuel.**
- Criticality safety: typically **to be ensured and demonstrated both in the operational and in the post-closure phase** of the DGR:

In the operational phase

- Limited time frame

  → direct controls/actions;
- Analogous measures for criticality control as implemented in nuclear facilities presently in operation.

In the post-closure phase

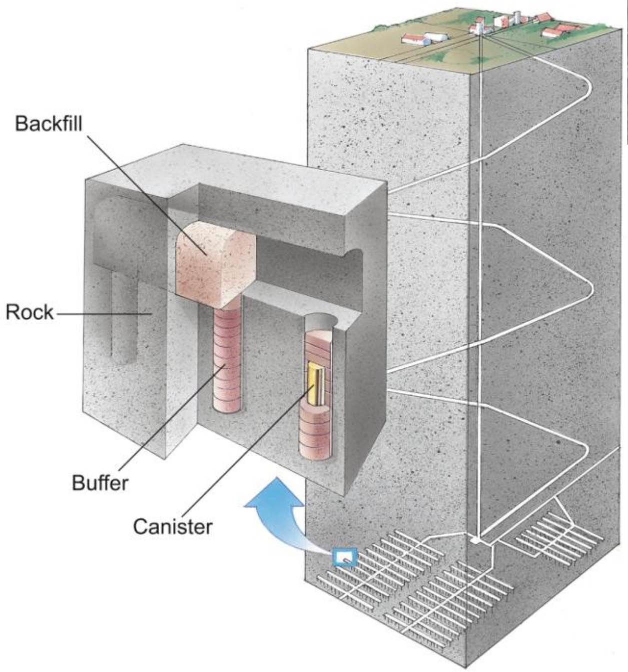
- Dedicated approach required
- Long time frames: orders of magnitude larger than in any other areas of the fuel cycle;
- Handling of uncertainties associated to the long-term evolution of the system.

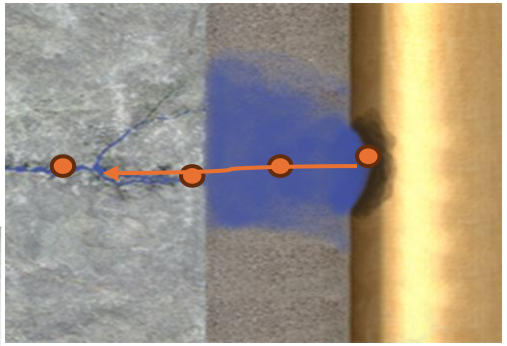
INTRODUCTION (VII) – WP17 CRITICALITY SAFETY FOR FINAL DISPOSAL (CSFD)

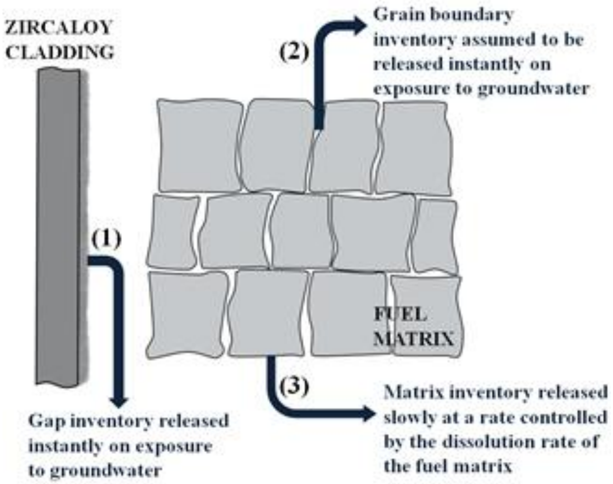
- The R&D work in WP-17 supports national programmes by addressing the challenges of ensuring and demonstrating post-closure criticality safety for long time scales, considering the variability of national DGR concepts and inventories.
- **The underpinning goal: consolidate the technical basis of the criticality safety argumentation for final disposal of fissile wastes.**
- This is achieved by:
  - Identifying and further exploring the optimisation potential of **measures for ensuring criticality** safety in final disposal → focus on the DGR post-closure phase:
    - Technical measures: e.g. optimising the design of final disposal containers for high-level waste;
    - Administrative measures: e.g. deriving fissile material limits per waste package (loading curves);
    - etc.
  - Further developing & improving the understanding of the various **methodologies to assess the effectiveness** of the measures considered.
    - Validation and experimental verification of approaches for criticality safety assessments.

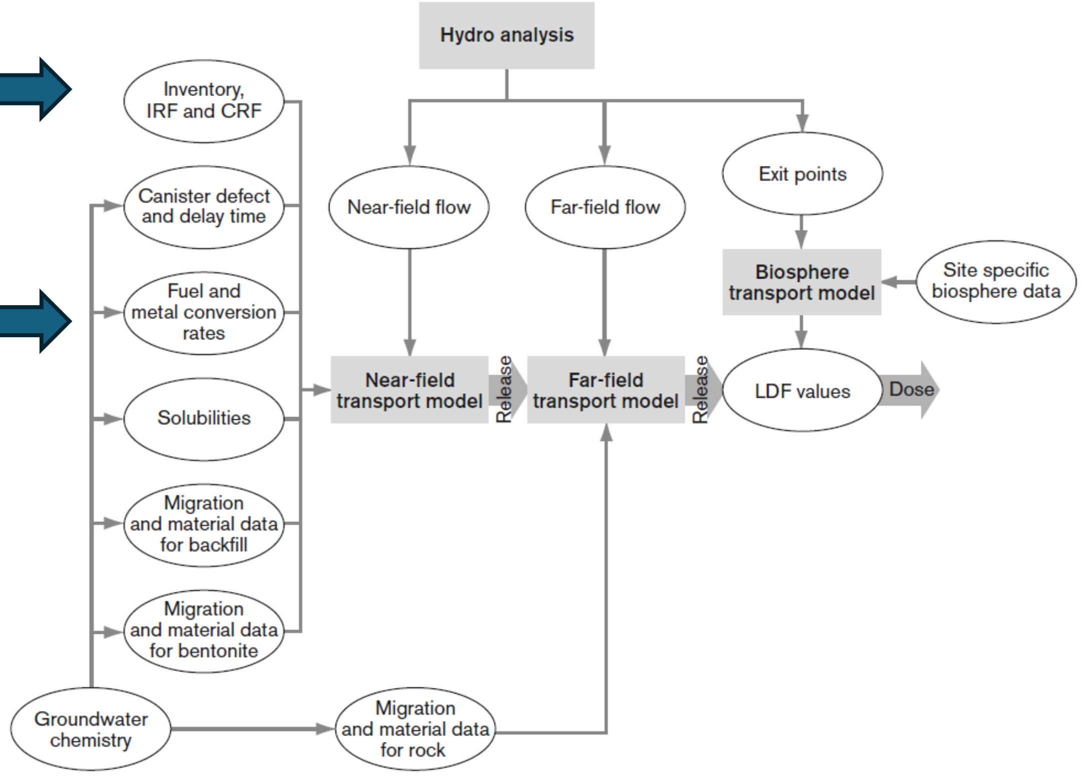
INTRODUCTION (IV) – RN RELEASE FROM A FINAL REPOSITORY

- Radionuclide transport in a consequence calculation when canister fails









INTRODUCTION (VI) – WP8 SAREC

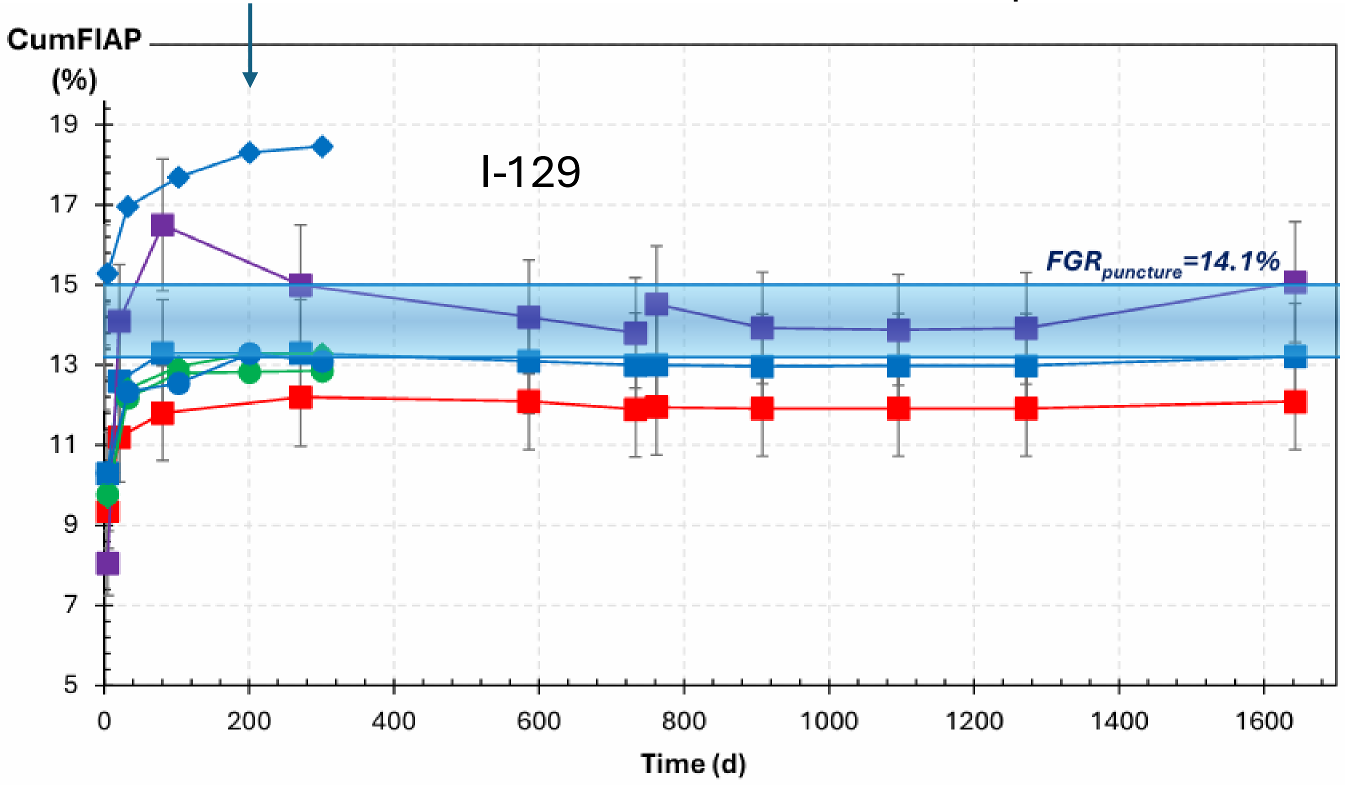
- R&D WP that will run for 5 years
- Our focus is **radionuclide release** from spent fuel: fission gas release, "instant release" and release via matrix dissolution. (I-129, Cs-127 etc).
- Work is divided in tasks:
  Spent fuel experiments in hot cell facilities
  Studies of **model materials**: UO_2 with or without dopants
    - Grain boundary focus
    - Microstructure
    - Chemical reactivity

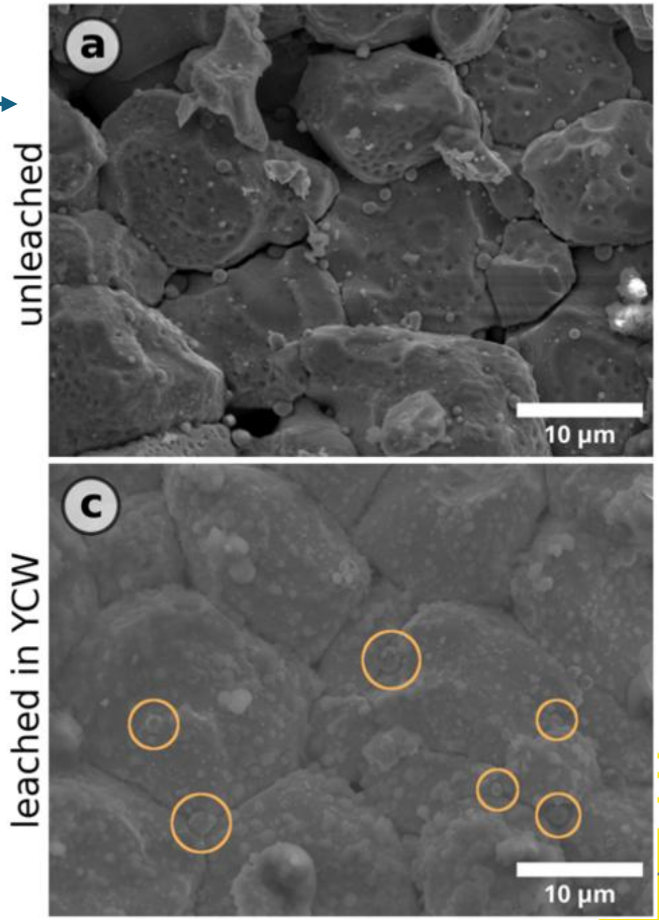
  Modelling via various approaches
- Knowledge management subtask is to develop a spent fuel leaching **database**
- Here: EXAMPLES of preliminary results available
  Will show some detailed graphs but will not describe in detail.

WP8 SAREC TECHNICAL RESULTS EXAMPLES (I)

No apparent metallic particle dissolution.
Secondary phases precipitates on surface in Young Cement Water

- **Spent Nuclear Fuel leaching (Task 3) – Few results available**
  Fuel selection, characterization and preparation : done .
  Leaching set-up & initiation ongoing.
  SCK CEN started before Eurad-2; **FZJ** post mortem analyses.

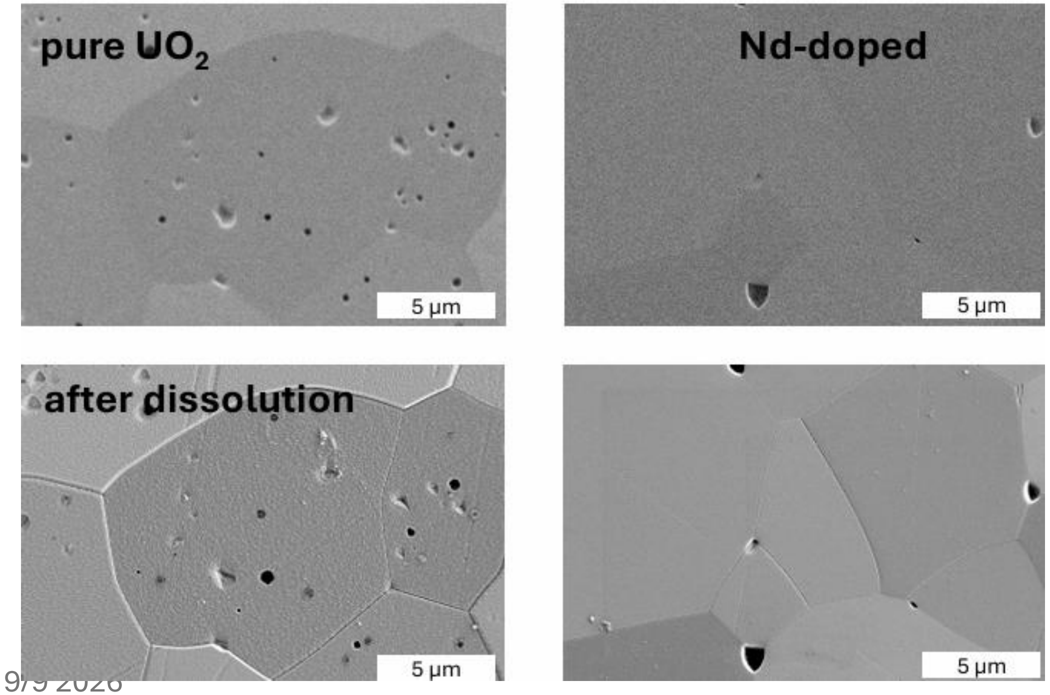




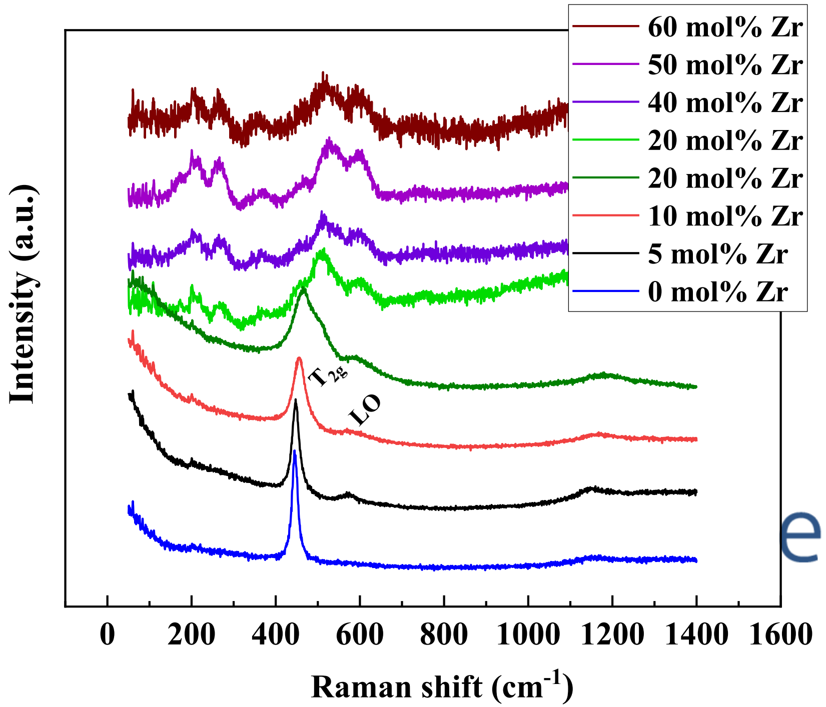
WP8 SAREC TECHNICAL RESULTS EXAMPLES (II)

- **Model materials (Task 4 & 5):**
  Materials synthesis - preparation & characterization.
  Few leaching results availabe.

FZJ show preferential grain boundary dissolution in La-doped UO2



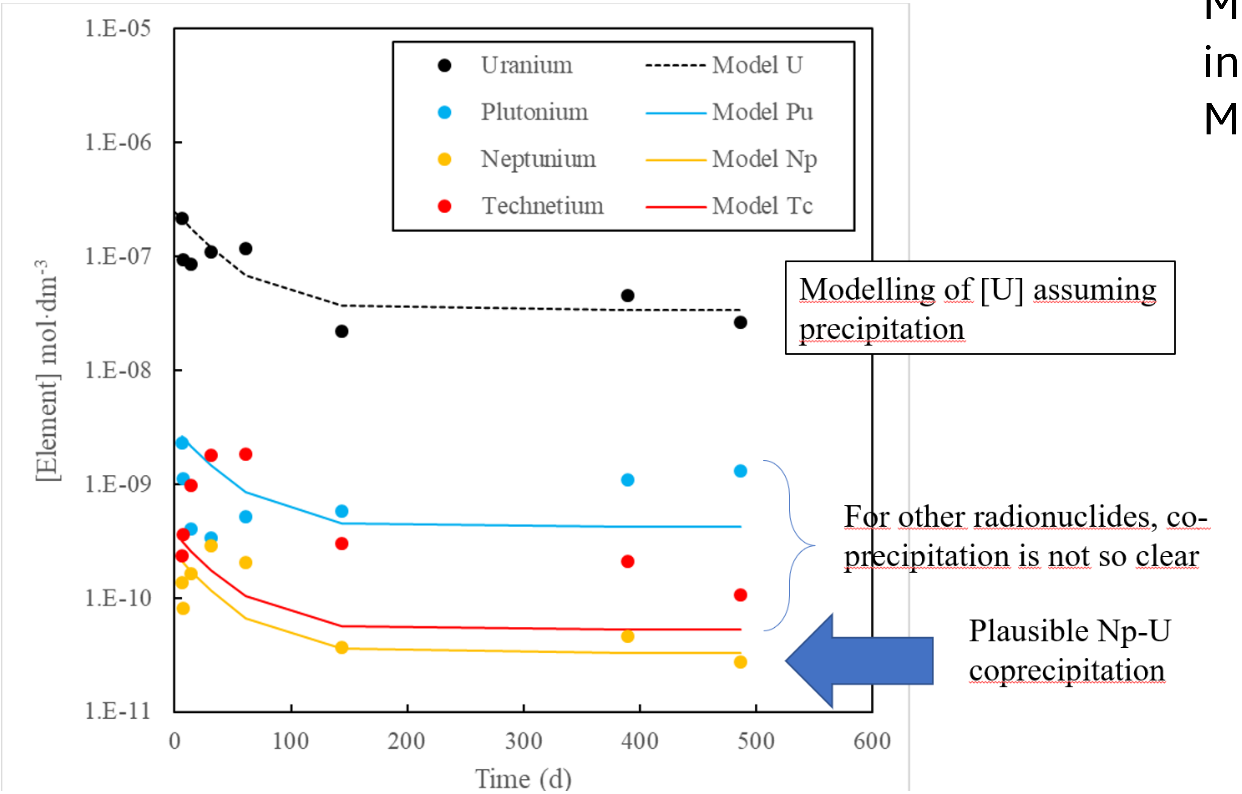
Ciemat: UO_2–ZrO_2 model materials
Coprecipitation & sol-gel (at JRC)
Raman shows Zr-induced structural transformation: dominant tetragonal structure at higher Zr contents



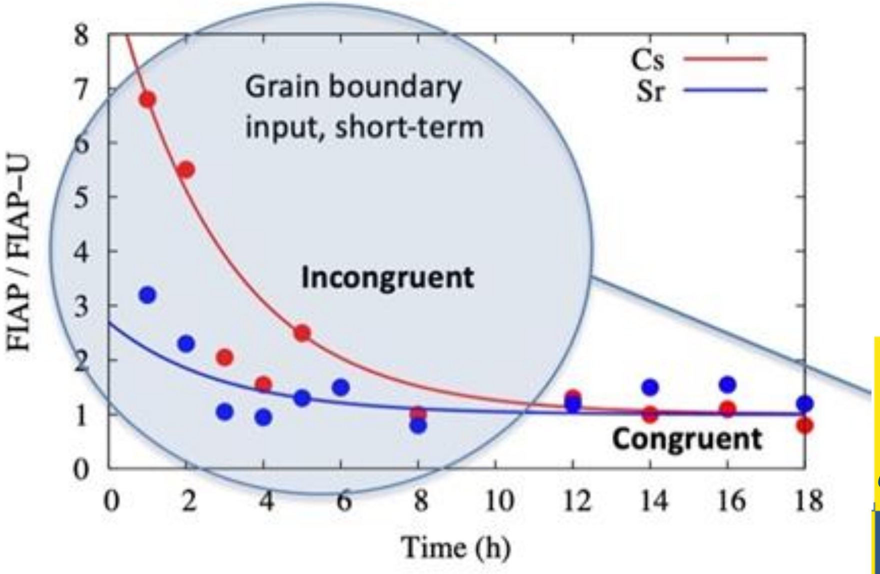
WP8 SAREC TECHNICAL RESULTS EXAMPLES (III)

- **Modelling Task 6.1 – various approaches to model fuel dissolution and radionuclide release**
  - Purpose to show and implement process understanding in models that are needed for the radionuclide transport models in the safety assessments.

UPC: semi-empiricial SERNIM; High pH leaching



Mines Paris: HYTEC & Thermochimie, modelling incongruent initial release of Cs & Sr.
Model for long term incongruence under development.



WP8 SAREC TECHNICAL RESULTS EXAMPLES (IV)

- **Modelling Task 6.2 – using fuel performance codes to model fission gas release (FGR) and Instant release fraction (IRF)**

PSI – Radionuclide inventory data from BEAVRS = Benchmark for Evaluation and Validation of Reactor Simulations using CASMO-5 & SIMULATE-5.
Resulting Isotopic database & max burnup ~33 MWd/kg

composition at /101202/10/ [isotopes.h5 in /afs/psi.ch/...

Table Import/Export Data

0-based

	isotope	values_eoc	values_decay	activity
0	H3	0.049999	0.047264	1.6825E13
1	C14	0.0	0.0	0.0
2	CL36	0.0	0.0	0.0
3	FE55	0.44545	0.34459	3.0728E13
4	CO60	0.0	0.0	0.0
5	NI59	8.839	8.8389	2.6453E10
6	NI63	1.6019	1.5908	3.3406E12
7	GE76	0.29683	0.29683	2.3545E-4
8	GE77	2.7392E-4	0.0	0.0
9	GE77M	5.1453E-7	0.0	0.0
10	GE78	1.9144E-4	0.0	0.0

Energorisk – FGR calculation using function of fuel burnup and linear power- needs burnup and linear power from calculations

$$FGR\,(\%) = A \cdot \left(\frac{BU}{BU_{ref}}\right)^{\alpha} \cdot \left(\frac{LP}{LP_{ref}}\right)^{\beta}$$

VTT – planned work:
a statistical IRF estimate using realistic reactor operating conditions derived from the benchmark specifications

TECHNICAL RESULTS WP8 SAREC (V)

- **SNF database (Task 2.3)**

A21 developed structure for the spent fuel leaching database – currently available as Excel in Project Place

Includes a plotting tool

Type of Data	Database ID	Author ID	Author&Year	Year Publication	Data Origin	Data Procedence	Reference
					• First Nuclides • DisCo • Eurad-2 • Other Sources	• Peer Reviewed • Project Proceedings • Technical Report • Doctoral Thesis	
Fuel	**Reactor** • PWR • BWR • LWR	**Fuel Type** • UOx • MOx • Al/Cr-UOx • Gd-UOx	**Original pellet geometry** • Diameter • Heigh	• **Enrichment (% U^{235} ; Pu^{239})** • **Irradiation cycles & time**	**BU (GWd/tHM) & Fuel Power (W/cm)** • Average • Local	**Cooling time**	• **Initial Inventory** • **Dose Rate (Gy/h)**
Sample	**Type** • Pellet • Segment • Fragment • Powder	**Region** • Whole • Center • Rim	**Cladding** • Yes / No • Cut • Detached	**Size of material (mm)**	**SA (m2/g)**	**Mass (g)** **Volume (L)**	**Position in Rod (mm from botton)**
Exp. Conditions	**Atmosphere** • Oxic • Anoxic • Reducing **Atm. Comp.** (%)	**Temp. (ºC)**	**Type of experiment** • Batch • Replenish. • Dynamic	**Solution type** • GW, BIC, Borate, NaCl, YCW-Ca, DIW	**Solution Comp. (mM)** • $NaHCO_3$, NaCl, Ca^{2+}, Borate, Br^-, F^-, SO_4^{2-}, PO_4^{3-}, SiO_3^{2-}	**Presence of Solid** • Fe(s)	**Pore size filtration** (colloid presence)
Exp. Results	**Accumulative Time (y)**	**Second. Phase precip.**	**FGR %**	**FIAP %**	**Concentration (in Molarity)**	**Concentration Uncertainties**	
FIAP %	**Rb**; Rb-85; Rb-87; **Cs**, Cs-137, Cs-133, Cs-134, Cs-135; **Sr**, Sr-90; **Ba**, Ba-138; **Mo**, Mo-97, Mo-98, Mo-100; **Tc**, Tc-99; **U**, U-238; **Np**, Np-237; **Pu**, Pu-239; **Cm**, Cm-244; **La**, La-139; **Ce**, Ce-140, Ce-144; **Pr**; Pr-141; **Nd**, Nd-144; **I**-127, I-129; **Eu**, Eu-153, Eu-154; **Se**-79, Se-80, Se-82; **C**-14; **Pd**-106, **Ru**, Ru-100, Ru-106; **Y**; **Zr**; **Rh**; **Am**, Am-241, Am-243; **Ag**, **Cd**, **Te**, **Pm**, **Sm**, **Sb**-125						

TECHNICAL RESULTS WP8 SAREC - SUMMARY (VI)

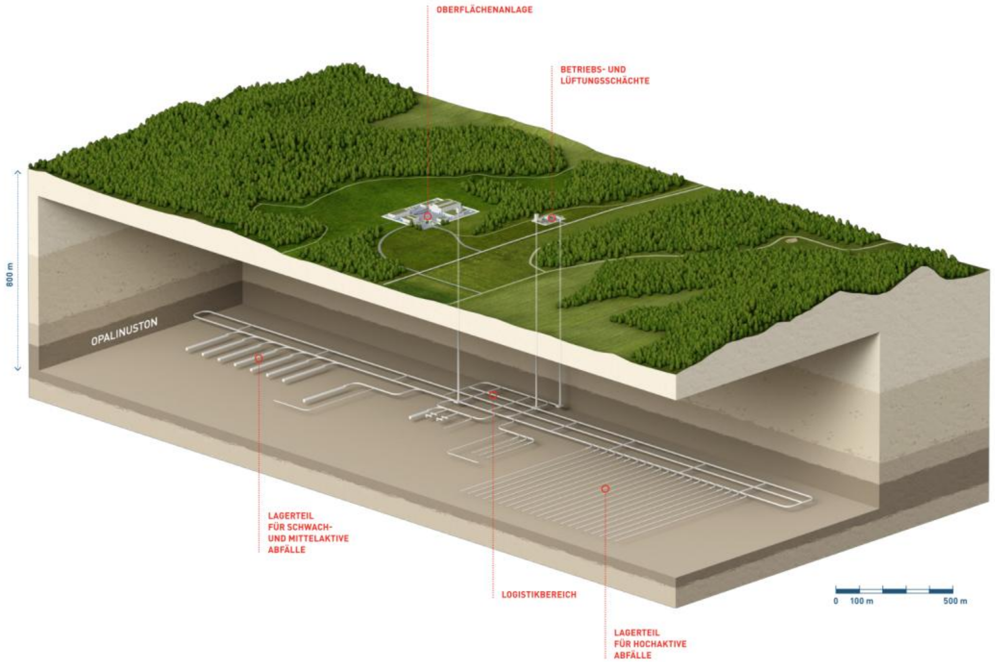
- **Task 2 – Knowledge managament**
  - Database for spent fuel leaching under development
- **Task 3 – Autoclave leaching experiments**
  - Most experiments started, early days but some results available
- **Task 4 & 5 – Grain boundary studies and leaching of model materials**
  - Materials being produced, only few initial leaching data available . One grain boundary study ready
- **Task 6 – Modelling**
  - Various approaches, …semi-empirical, kinetics, thermodynamics, fuel performance codes … discussions on input – output data held

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (I)

- WP17 "CSFD" addresses the unique challenges in ensuring and demonstrating post-closure criticality safety posed by the extremely long time frames required for assessments:

  - We are **investigating & designing measures**
  - We are developing and **exploring new approaches to assessing the DGR safety performance**

For a system that is evolving continuously over hundreds of thousands of years.



Pictures courtesy of Nagra.

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 3)

- **Task 3** is consolidating the understanding of the system post-closure evolution

- **Task 3 is a "translator" of phenomenological evolutions into criticality relevant parameters to help defining modelling assumptions**

- Milestone: carrying out a survey and developing a **catalogue of post-closure features, events and processes** (FEP) that, for the very first time, have a **specific focus on post-closure criticality safety**.

2 - External factors

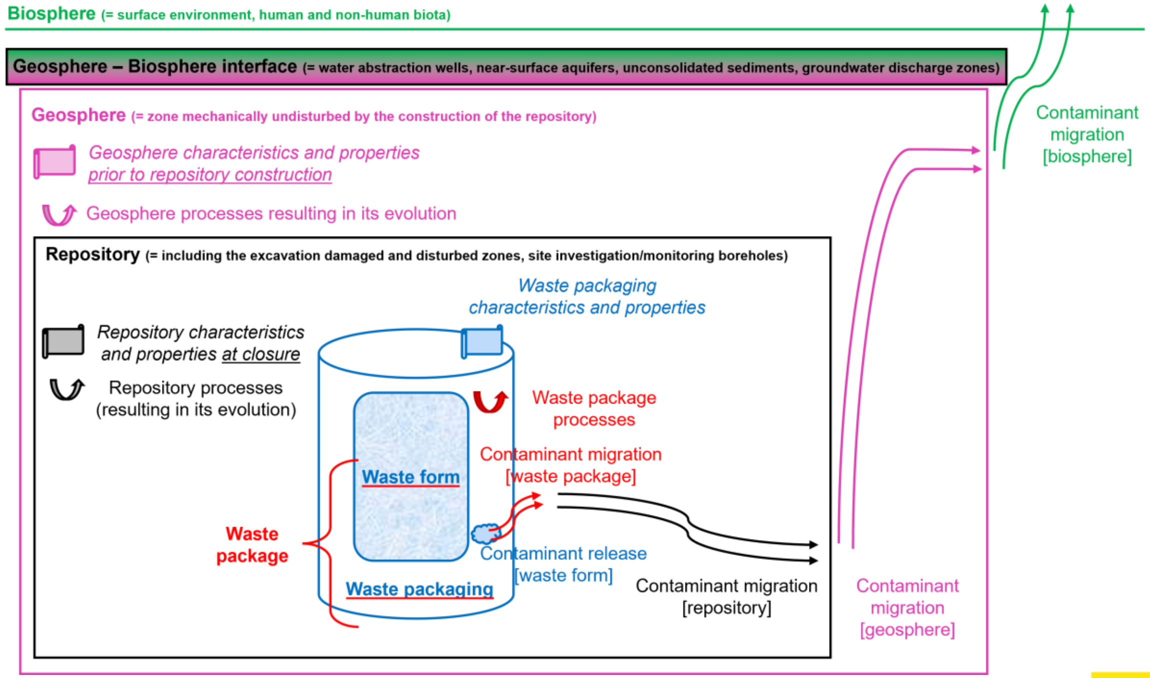


Illustration of the considered FEP categories.
Courtesy of Andra

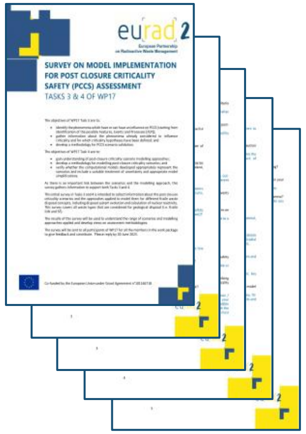
TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 4)

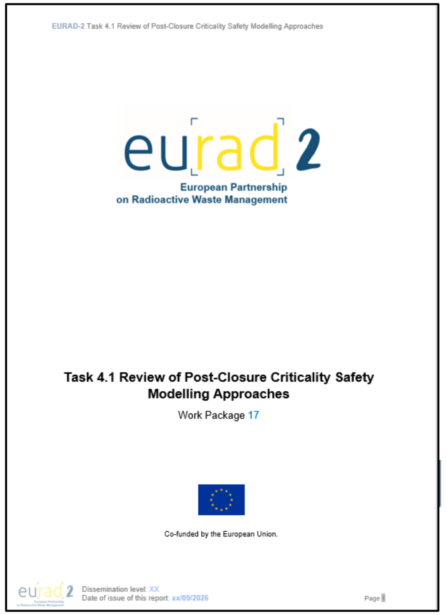
- **Task 4** is investigating modelling approaches for post-closure criticality safety assessments that address both the need for detailed computational models and the extremely large parameter space.

- **Objective: Review PCCS scenario modelling approaches for different waste disposal concepts**

- **Questionnaire: current practices for post-closure criticality safety scenarios & modelling approaches**
  - How has the regulatory context influenced the modelling approach and assumptions?
  - What scenarios are considered? How are they evaluated?
  - How is the disposal system evolution assessed or modelled to determine relevant wasteform and barrier properties at various times after disposal?
  - How are the criticality configurations of evolved conditions specified for evaluation? Deterministic representations of conditions or some form of coupled modelling of disposal system evolution and nuclear reactivity?
  - How are uncertainties treated? How are models simplified?

- **Task 4.1 report**
  - Collates questionnaire responses, identifying commonalities and differences between national approaches to inform Tasks 3 and 4
  - **To be published shortly**




EURAD-2 Task 4.1 Review of Post-Closure Criticality Safety Modelling Approaches

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Task 4.1 Review of Post-Closure Criticality Safety Modelling Approaches

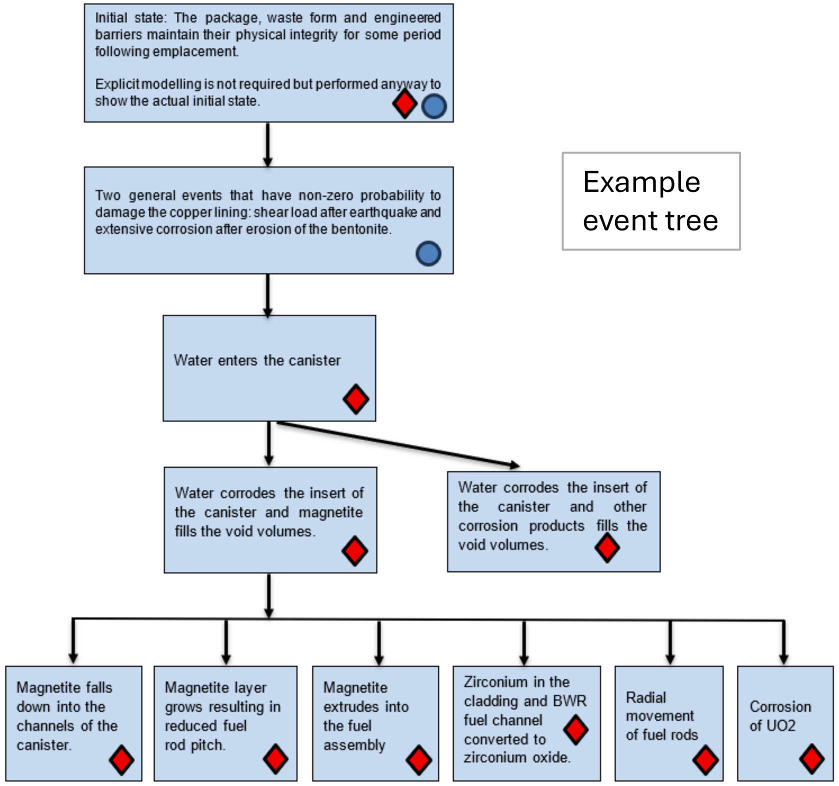
Work Package 17

Co-funded by the European Union.

Dissemination level: XX
Date of issue of this report: xx/09/2026
Page

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 4 II)

- **Task 4.2: Proposed approach in two stages**
  - Start position – understand the disposal system and its evolution => criticality scenarios
  - End position – have a base neutron transport model for each system/scenario & a list of scenario uncertainties and sensitivities to analyse in Task 4.3
- **Stage 1 – Methodology (ongoing):**
  - Develop event trees summarising disposal system evolution and identify criticality safety methodology for criticality scenarios
  - For each system state identify whether criticality can be excluded using qualitative arguments, bounded by another scenario or explicit neutron transport modelling is required
  - If qualitative arguments can exclude criticality, what arguments and evidence are applied?
  - If modelling is required, define the conceptual model - what assumptions and simplifications would be made? Are these justified/supported by evidence? Would conservative or best estimate parameter values be used? What uncertainties and sensitivities need to be considered?
- **Stage 2 – Model Development (from this autumn):**
  - Each participant to prioritise what criticality scenario uncertainties and simplifications they want to undertake sensitivity analysis for, followed by collation and group review to prioritise and avoid duplication
  - Develop a base model for the prioritised criticality scenarios to be investigated
  - Cross-compare base model results between participants where applicable



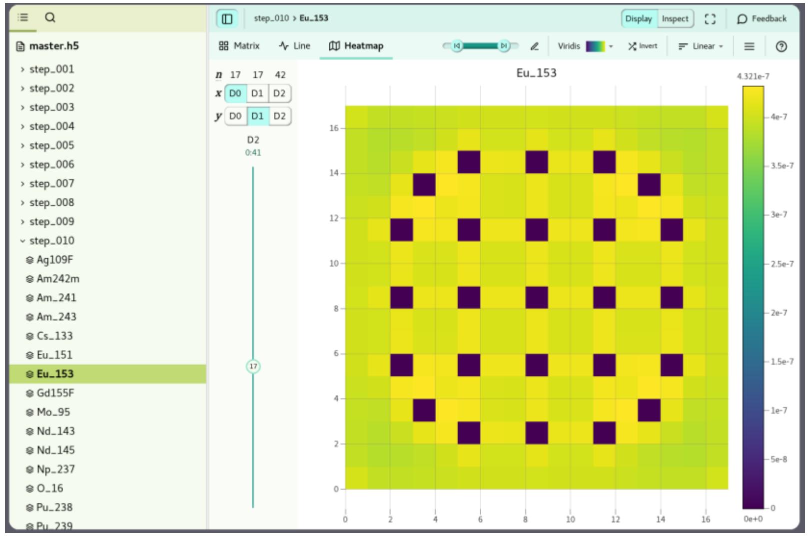
Example event tree

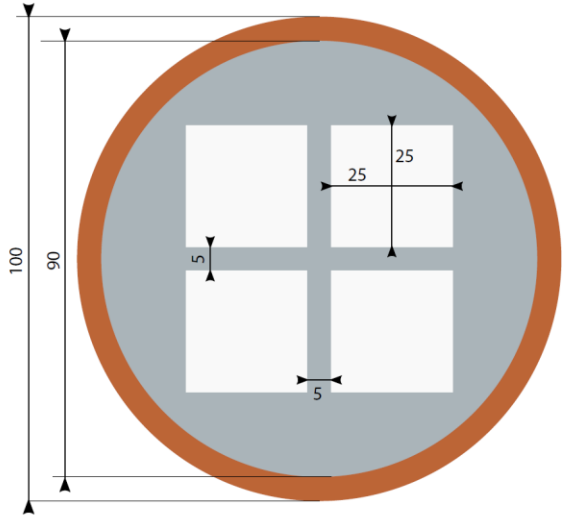
Figure: Event tree for the generic post-closure criticality safety assessment of KBS-3 containing SNF in bedrock.

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 5)

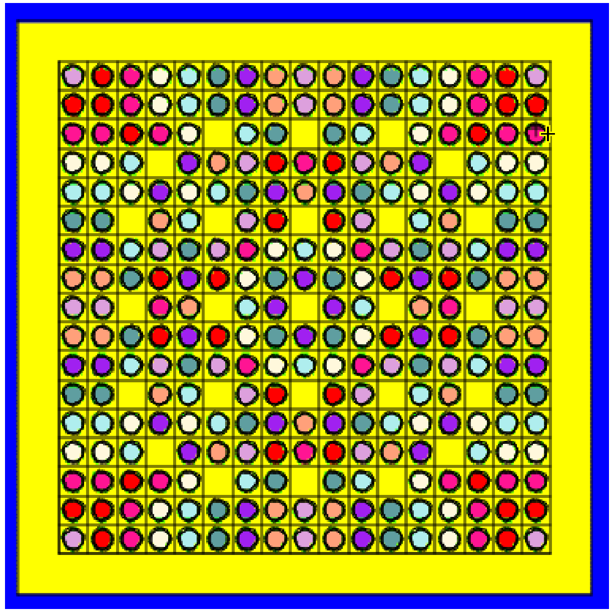
- Task 5 is investigating and further developing:
  - Approaches to deriving limits to the fissile material amount per waste package;
  - The factors that influence the derivation of fissile material limits with a view to optimise waste package & barrier design.
- **Current focus**:
- Task 5.1: loading curve derivation → ref. canister with burnt FAs
  - 17x17-pins FA with 2% to 5wt.%
  - Burnt with PWR representative conditions
  - 4 identical burnt FA in canister
- Task 5.3: canister sensitivity studies with fresh FAs
  - Steel corrosion (1 to 5cm)
  - Fuel assembly shift (-1.79cm to 1.79cm)
  - Space between neighboring fuel assembly (0 to 10cm)
  - Water density (70% to 102%)

Benchmark





TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 5 II)



- **Task 5.1 benchmark definition**:
  - Burnt fuel composition shared
  - Fuel assemblies modelled with 40 axial levels
  - 1/8 radial symmetry
  - ENDF/B-VII.1 nuclear data library as reference

- About 15 organisations participate

- Multiple Monte Carlo neutron transport codes:
  - Serpent-2
  - Scale
  - MCNP
  - OpenMC

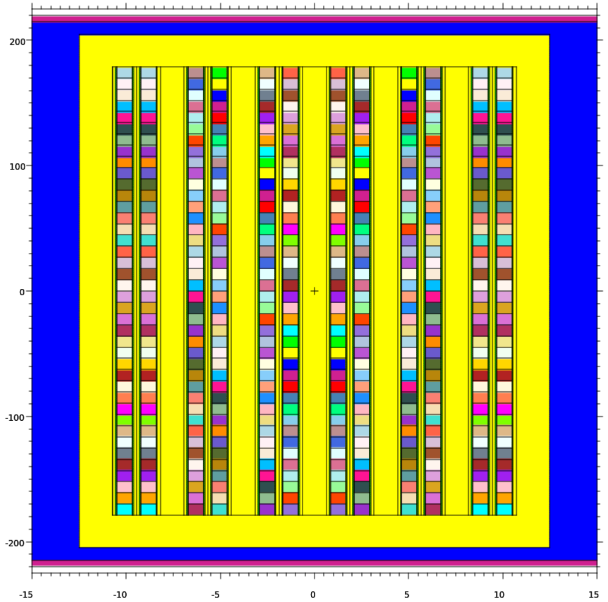


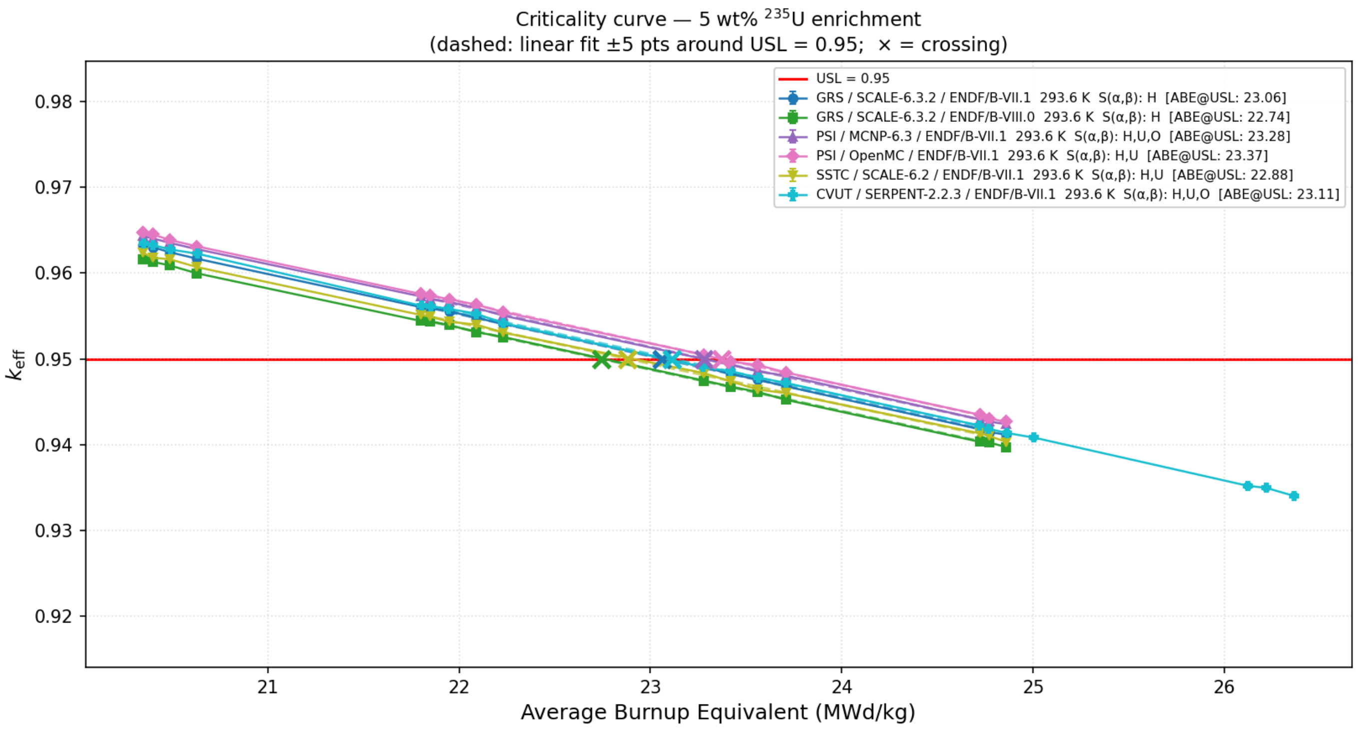
Illustration of an MCNP model: axial view (left) & transverse section (above)

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 5 III)

- **Task 5.1 preliminary results**

Org.	Code	Lib	3wt%	4wt%	5wt%
GRS	SCALE-6.3.2	B-VII.1	3.92	13.63	23.06
GRS	SCALE-6.3.2	B-VIII.0	3.71	13.37	22.74
PSI	MCNP-6.3	B-VII.1	3.96	13.76	23.28
PSI	OpenMC	B-VII.1	4.08	13.86	23.37
SSTC	SCALE-6.2	B-VII.1	4.17	13.51	22.88
CVUT	SERPENT-2.2.3	B-VII.1	3.94	13.66	23.11
Mean			3.96	13.63	23.07
Std dev			0.16	0.17	0.24

Table 1: Average burnup equivalent in MWd/kg at USL=0.95



TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 5 IV)

- **Task 5.3 objective:**
  canister sensitivity studies with fresh FAs

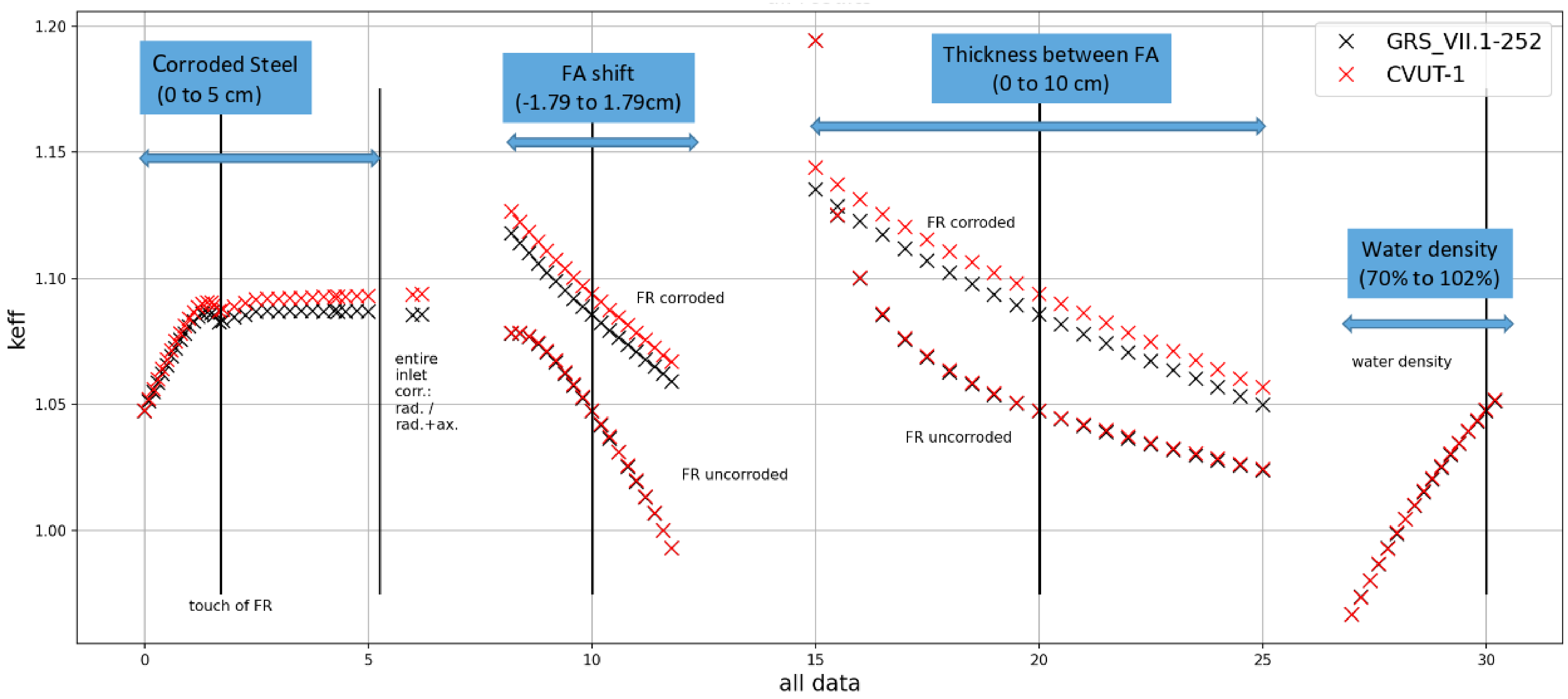
- **Task 5.3 benchmark definition**:
  - Corrosion of stainless-steel inlet as magnetite:
    - Pilling-Bedworth ratio of 2
    - Formation of MnO_2, Cr_2O_3, NiO and Fe_3O_4 compounds
    - Magnetite density of 5.5072 g/cm^3
      (i.e., 6 % higher than pure magnetite)

Overview of parameters to be varied in sensitivity study

ID	Description	Comment
1.1-1.3	**Corrosion of SS inlet** 0-1.5cm: water displacement 1.564cm: water contact FA 1.6788cm: water contact fuel pin	SS replaced by magnetite Corrosion rate: 2um/year
2.1-2.5	**Corrosion of SS inlet** 1.75cm-4.25cm: water gap above/below FAs filled with magnetite 4.3198 cm:magnetite touch copper coating 4.5cm, 4.75cm and 5.0cm	Not physical
3.1-3.3	**Corrosion of entire SS inlet** radially / radially+axially / between fuel rods	Not physical
4.1/4.2	**FA shift** (-1.79cm to 1.79cm) uncorroded/corroded canister	-
5.1/5.2	**Thickness between FA** (0 to 10cm) uncorroded/corroded canister	Box size stays the same
6	**Water denstity variation** (70% to 102%)	Optional

TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 5 IV)

- Task 5.3 objective: canister sensitivity studies with fresh FAs
- **Preliminary results**



TECHNICAL RESULTS WP17 CSFD – HIGHLIGHTS (TASK 6, 7 & 2)

- **Task 6** aims at identifying the experimental data required for post-closure criticality safety assessments and the types of experimental programme that could be done to address any significant gaps and uncertainties:
  - **Survey was carried out** to identify available experimental data and gaps where new data is required to meet PCCS assessment needs.
  - Report summarising the conclusions of the survey to be published in January '27.

- **Task 7** is exploring the potential of assessing the effects of potential post-closure criticality events with a view to improving the system understanding and demonstrating the robustness of the disposal concept
  - Task definitions were circulated.
  - **Phenomenology work** ongoing→ See also WP17 "Cutting-Edge Science" presentation on Thursday.

- **Task 2** aims at developing an effective communication strategy to all relevant stakeholders (general public, national regulator, etc):
  - First draft of **glossary** comprising technical terms regarding post-closure criticality safety is currently being reviewed.

HIGHLIGHT ON JOINT ISSUES

- (14:45) Over to Thierry

END USERS PERSPECTIVE

- (15:00) Over to Miguel

DISCUSSION WITH AUDIENCE

- **Any questions regarding the presentations?**

- **General discussion – for example**
  - What types of uncertainties are we faced with?

  - Using conservatism to handle uncertainties
    - Excessive conservatism
    - Handling of processes