

Development of depletion models for radionuclide inventory, decay heat and source term estimation in discharged fuel

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Outline

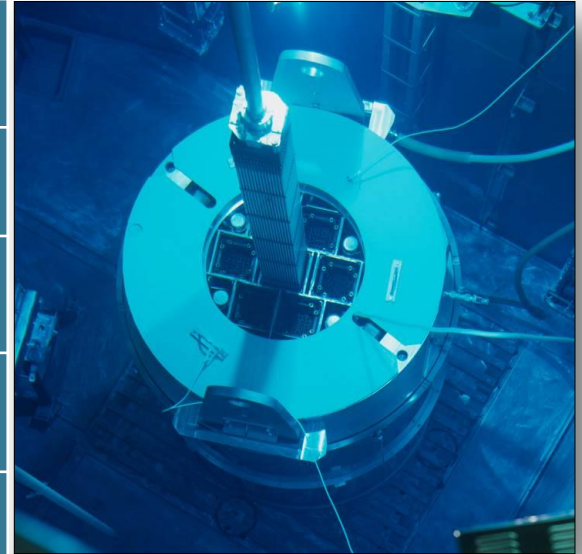
I. Spent fuel in Switzerland

II. Fuel depletion and cladding activation models

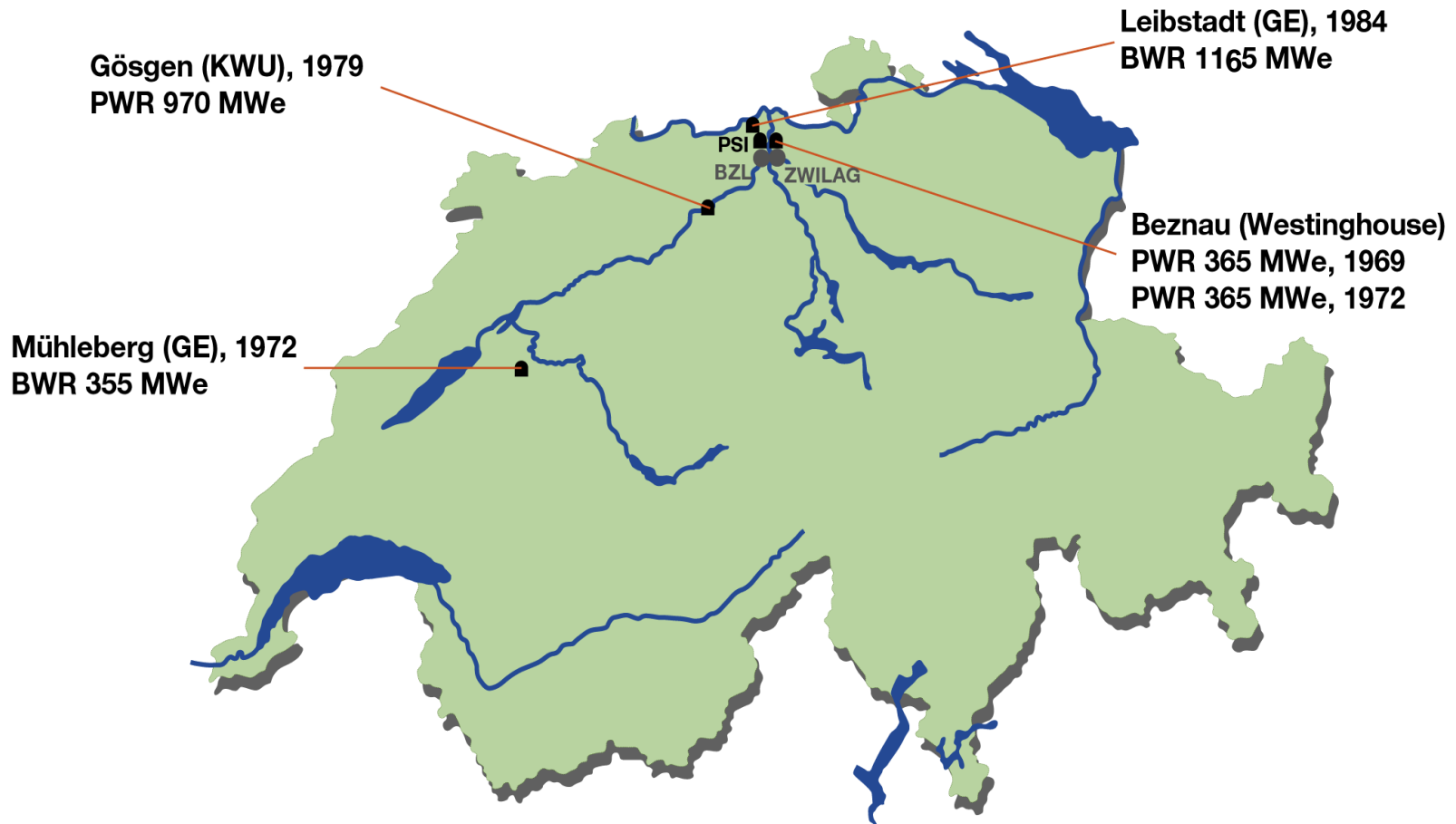
III. Model validation

IV. Applications

V. Conclusion and further steps



I. Swiss nuclear power plants



- 5 NPPs – installed output: 3,200 MW
- Percentage of total power production: ca. 40 %

I. Spent Fuel and Radioactive Wastes

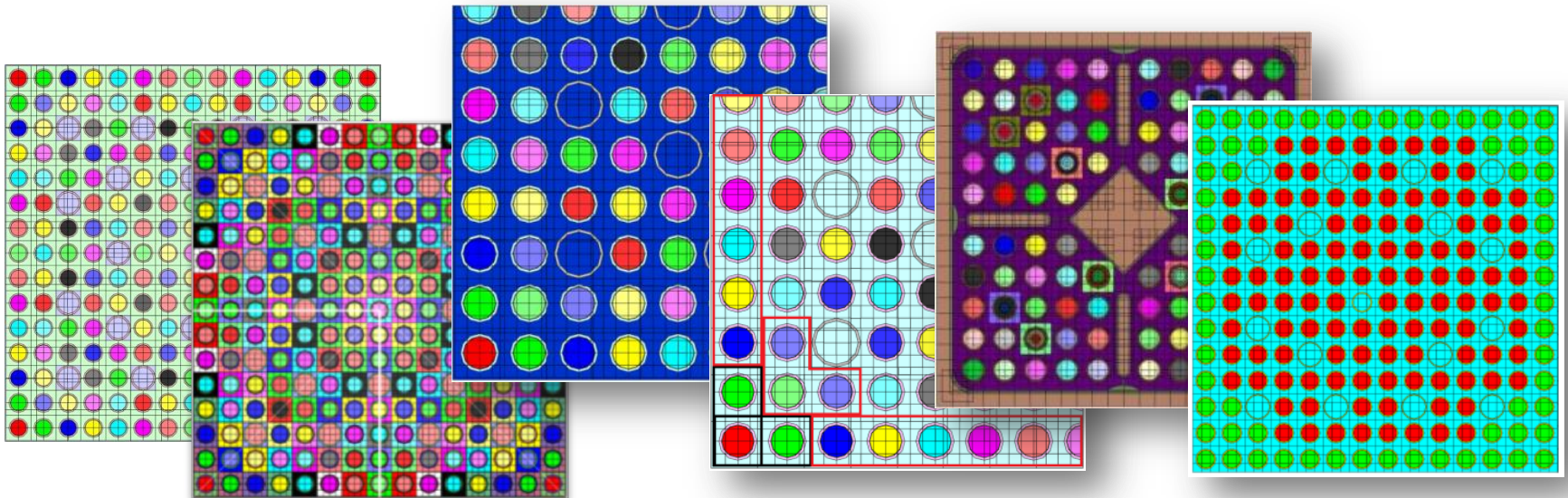
- SNF: ca. 12'000 spent fuel assemblies are expected to be discharged from the operation of the Swiss reactors (more than 2'000 FAs are currently in interim dry storage sites: ZWILAG, ZWIBEZ). Others in wet storage pools.
- For each of the Swiss reactors, different designs of fuel assemblies have been used and, among these, the fuel enrichment and burnup also vary. Furthermore, many of these spent fuel assemblies exhibit **very high burnup**

NPP	NPP type	Fuel type	FAs EOL
Beznau I	PWR	UO ₂	> 1500
Beznau II	PWR	MOX	ca. 230
Gösgen	PWR	UO ₂	> 1500
Gösgen	PWR	MOX	ca. 150
Leibstadt	BWR	UO ₂	> 7000
Mühleberg	BWR	UO ₂	> 1000
Tot UO ₂		UO ₂	> 10000
Tot MOX		MOX	ca. 380
Tot PWR		-	30%
Tot BWR		-	70%
Total			ca. 12000

- SNFs (and also HLW) are foreseen for deep geological disposal

II. Depletion Calculations

- **Scope of the study:** use of a state-of-the-art reactor physics code to model different types of fuel bundle, representative of the fuels employed in the Swiss power plants.
- I. **Calculation and validation** of the nuclide inventory against chemical measurements (samples from international projects: Malibu, Ariane, etc.)
- II. **Development** of binary libraries ready to be implemented in reactor physics code (e.g. Origen-ARP, Origami)
- III. **Estimation** of the full radionuclide inventory based on the new models.



II. Method for fuel depletion and cladding activation

- I. Development of a Fuel Assembly model with SCALE/Triton. Fuel depletion (POWER mode) and cladding activation (FLUX mode)
- II. Validation of calculated nuclides concentrations against available experimental data (work used also for T/S cask license application)
- III. Fuel libraries for ORIGEN-ARP and structure materials libraries for ORIGEN-S are generated (covering different fuel enrichment and burnup ranges)

In general ORIGEN-ARP is employed in Nagra-projects for :

- Determination of the radionuclides inventory of spent fuel for the MIRAM inventory (long term safety analysis)
- Determination of neutron and gamma source terms
- Decay power (canister loading optimization and long term SA)

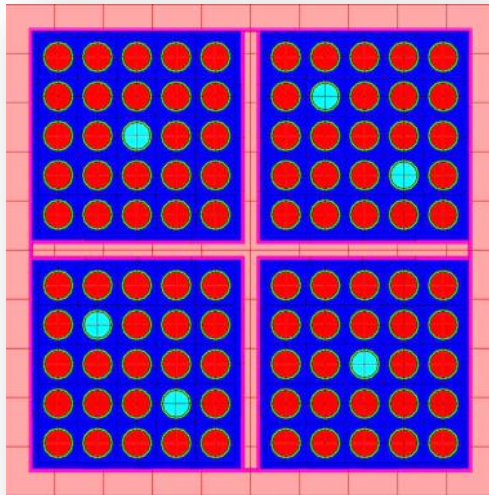
About the TRITON model and related Assumptions:

- TRITON is a sequence of SCALE (SCALE 6.1.3)
- Cross-section processing with CENTRM/PCM (ENDF/B-VII)
- Approximations: 1/8 symmetry of the FA, 1/4 FA model using mirror boundary conditions
- Optimized material numbers implemented - assign function
- No library collapse (no weight function)

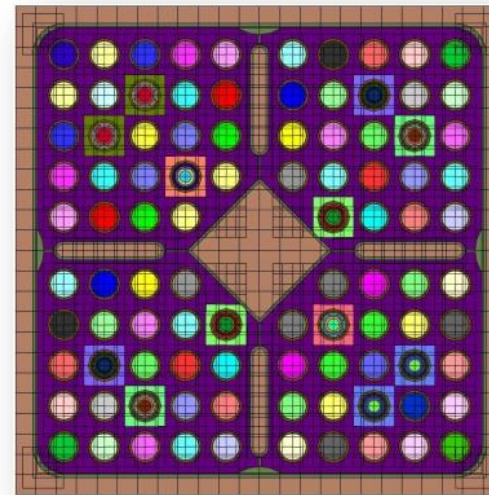
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ORIGEN-ARP standard
(SVEA 100)



ORIGEN-ARP developed
(SVEA 96)

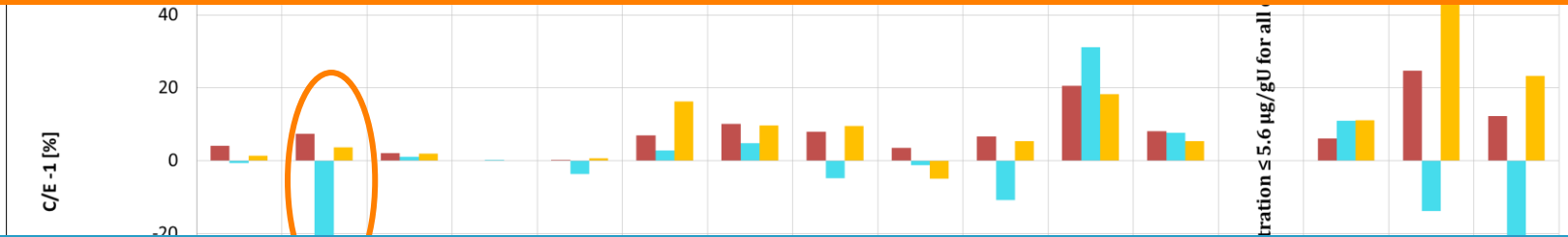


Reference: M. Gutierrez “Development and validation of 2D and 3D SCALE LWR fuel assembly models for burnup calculations and activation studies”, M.Sc. Thesis in Nuclear Engineering ETHZ (2015)

III. Model validation: SVEA-96 BWR (70GWd/t) 4.5 wt.% enr.

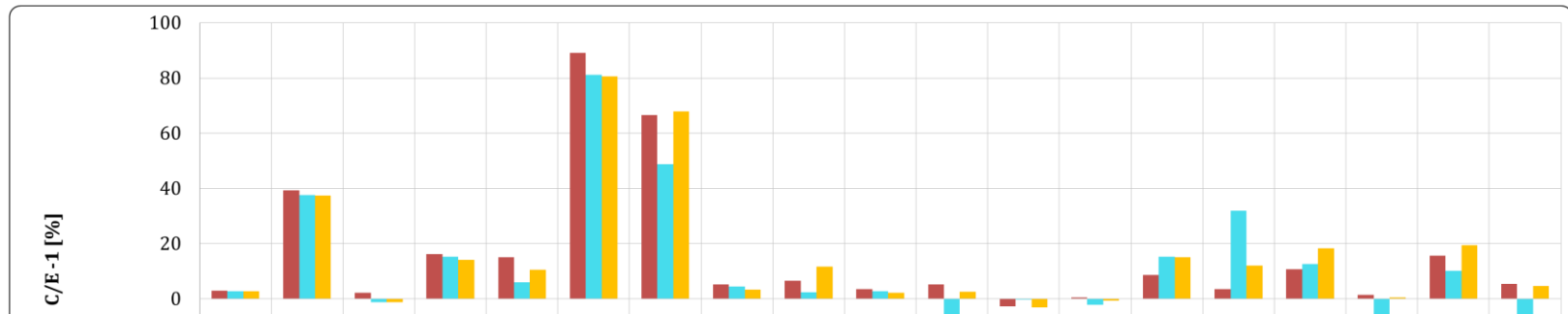
- General good C/E agreement

Better agreement ARP-96 compared to ARP-100: ^{235}U (3.7% vs -27.8%)



Worse agreement observed for the Pu vector (but within 20%)

	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	Am-242m	Am-243	Cm-243*	Cm-244	Cm-245	Cm-246
TRITON (sample)	+4.0	+7.4	+2.0	-0.0	+0.0	+6.9	+10.1	+8.0	+3.5	+6.7	+20.6	+8.0	+0.0	+6.0	+24.7	+12.3
ARP SVEA-100	-0.7	-27.8	+1.1	+0.1	-3.7	+2.8	+4.7	-4.8	-1.2	-10.8	+31.1	+7.6	+0.0	+10.9	-13.8	-21.2
ARP SVEA-96	+1.3	+3.7	+1.9	-0.0	+0.6	+16.2	+9.7	+9.5	-4.9	+5.4	+18.2	+5.4	+0.0	+11.1	+49.0	+23.3

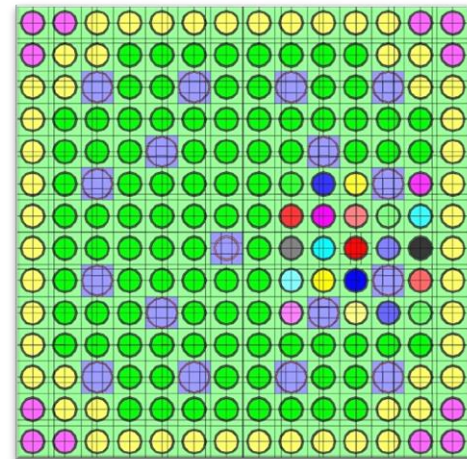
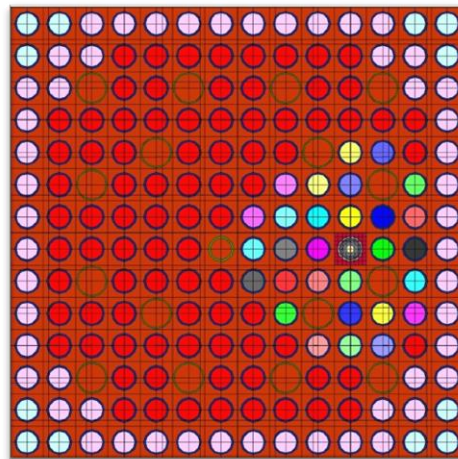
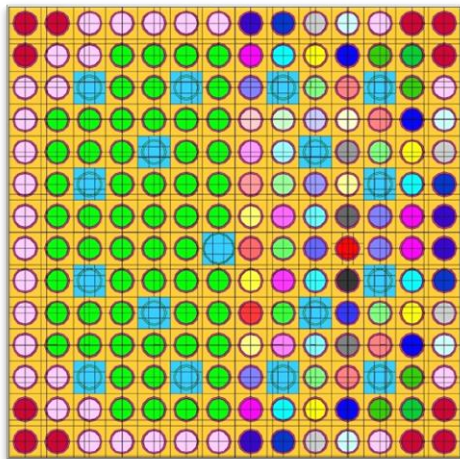


Fission products very well predicted

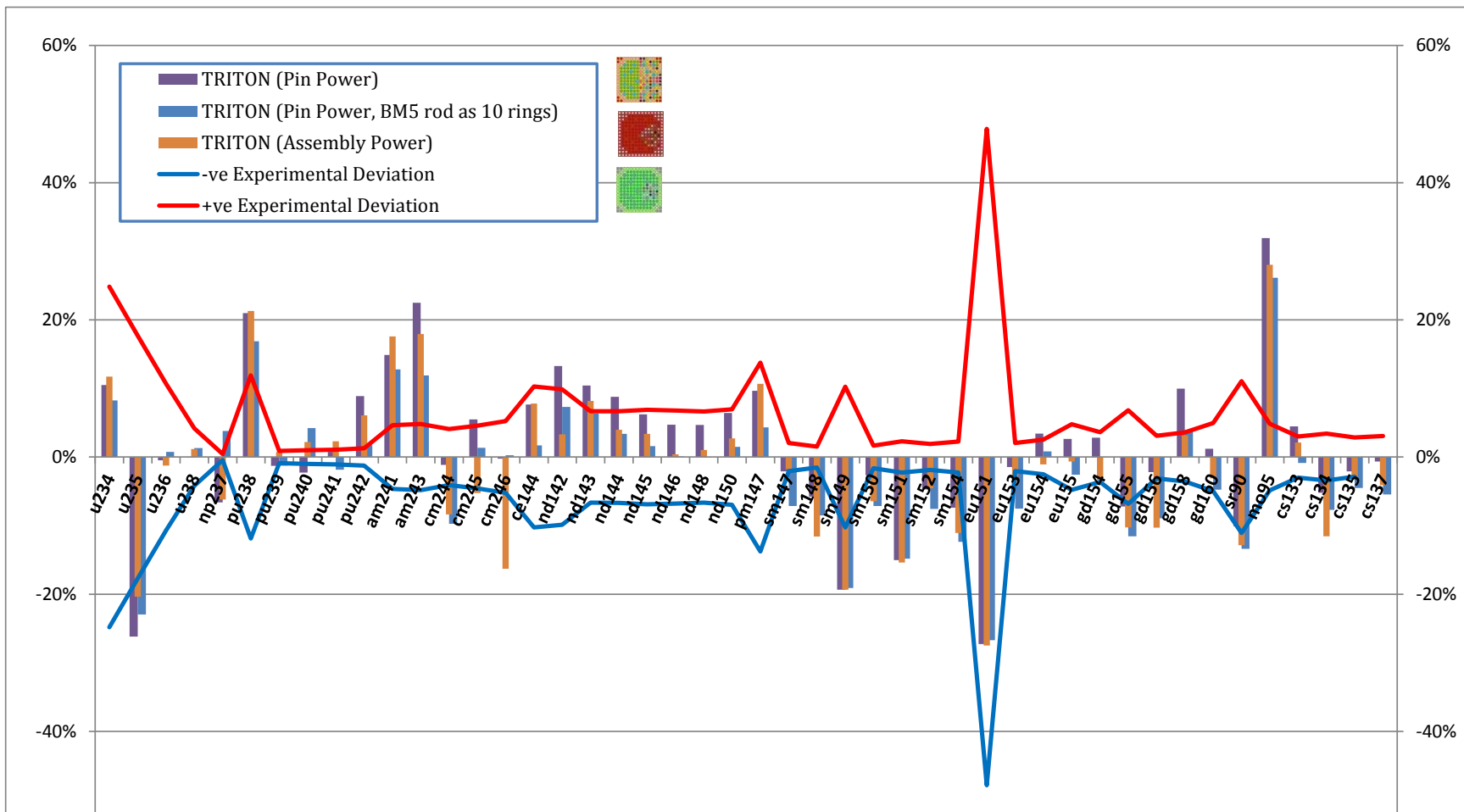
	Sr-90	Mo-95	Tc-99	Ru-101	Rh-103	Ag-109	Sb-125	Cs-133	Cs-134	Cs-137	Nd-143	Nd-144	Nd-148	Sm-149	Sm-151	Eu-154	Eu-155	Gd-154	Gd-155
TRITON (sample)	+3.0	+39.3	+2.3	+16.2	+15.0	+89.1	+66.6	+5.3	+6.6	+3.4	+5.2	-2.8	+0.5	+8.6	+3.5	+10.7	+1.5	+15.6	+5.4
ARP SVEA-100	+2.7	+37.6	-1.1	+15.3	+5.9	+81.2	+48.7	+4.4	+2.3	+2.8	-5.7	-0.2	-2.2	+15.2	+31.9	+12.6	-27.2	+10.1	-22.8
ARP SVEA-96	+2.7	+37.4	-1.2	+14.1	+10.5	+80.5	+67.8	+3.3	+11.7	+2.2	+2.7	-3.2	-0.7	+15.1	+12.1	+18.3	+0.5	+19.4	+4.7

III. Model validation: 14x14 MOX PWR (52Gwd/t) – ARIANE BM5

- TRITON in **pin power mode**: RHS rods are assigned/treated with individual material numbers.
- TRITON in **pin power mode**: fuel pin in ring-wise representation. Only rods neighbour to sample are assigned/treated with individual material numbers.
- TRITON in **assembly power mode**: only rods neighbour to sample are assigned/treated with individual material numbers.

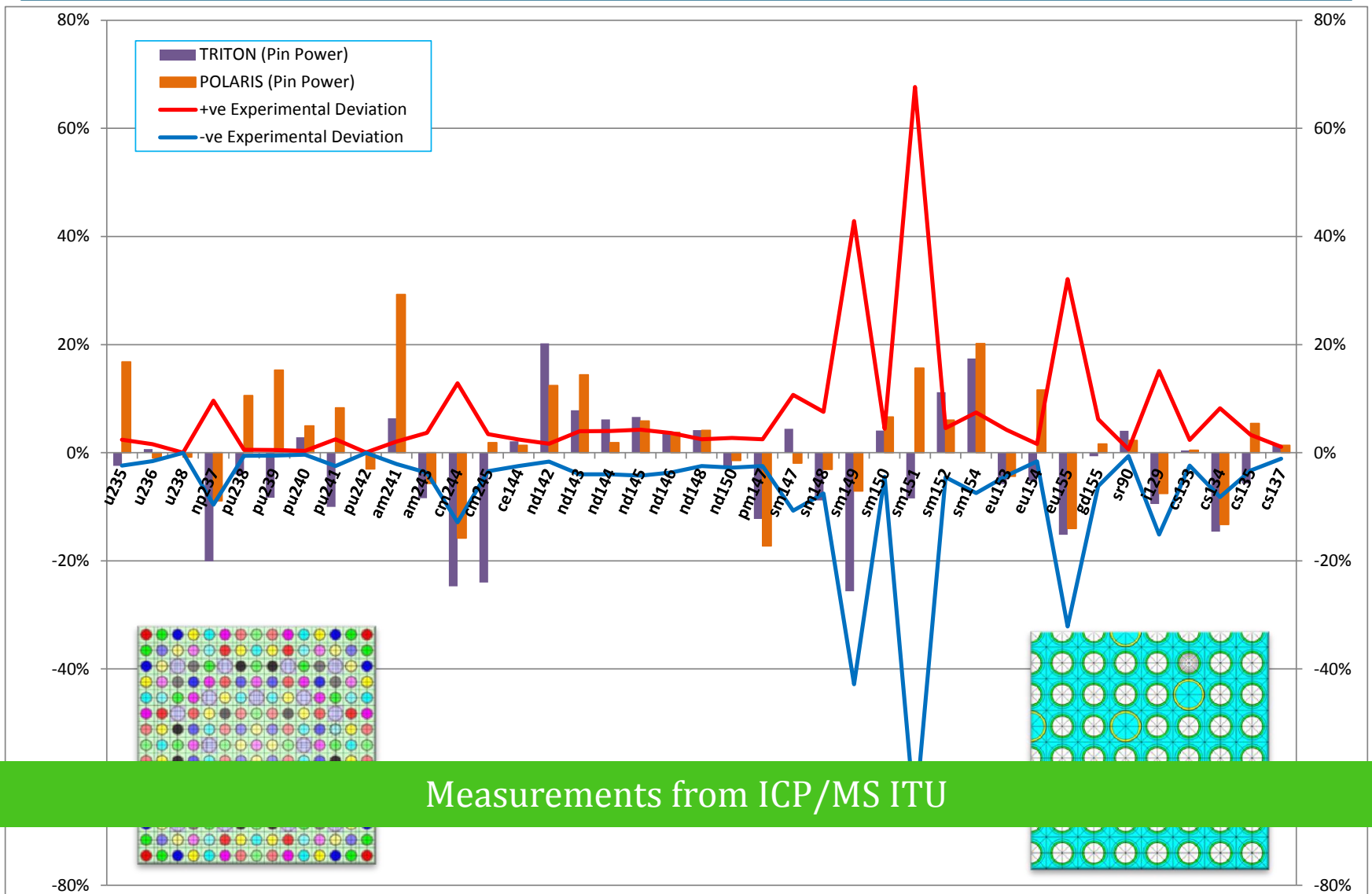


III. Model validation: 14x14 MOX PWR (52Gwd/t) – ARIANE BM5



Measurements from ICP/MS PSI

III. Model validation: 15x15 UO2 PWR (53Gwd/t) – ARIANE GU3



Measurements from ICP/MS ITU

IV. Application: Radionuclide inventory for spent fuel

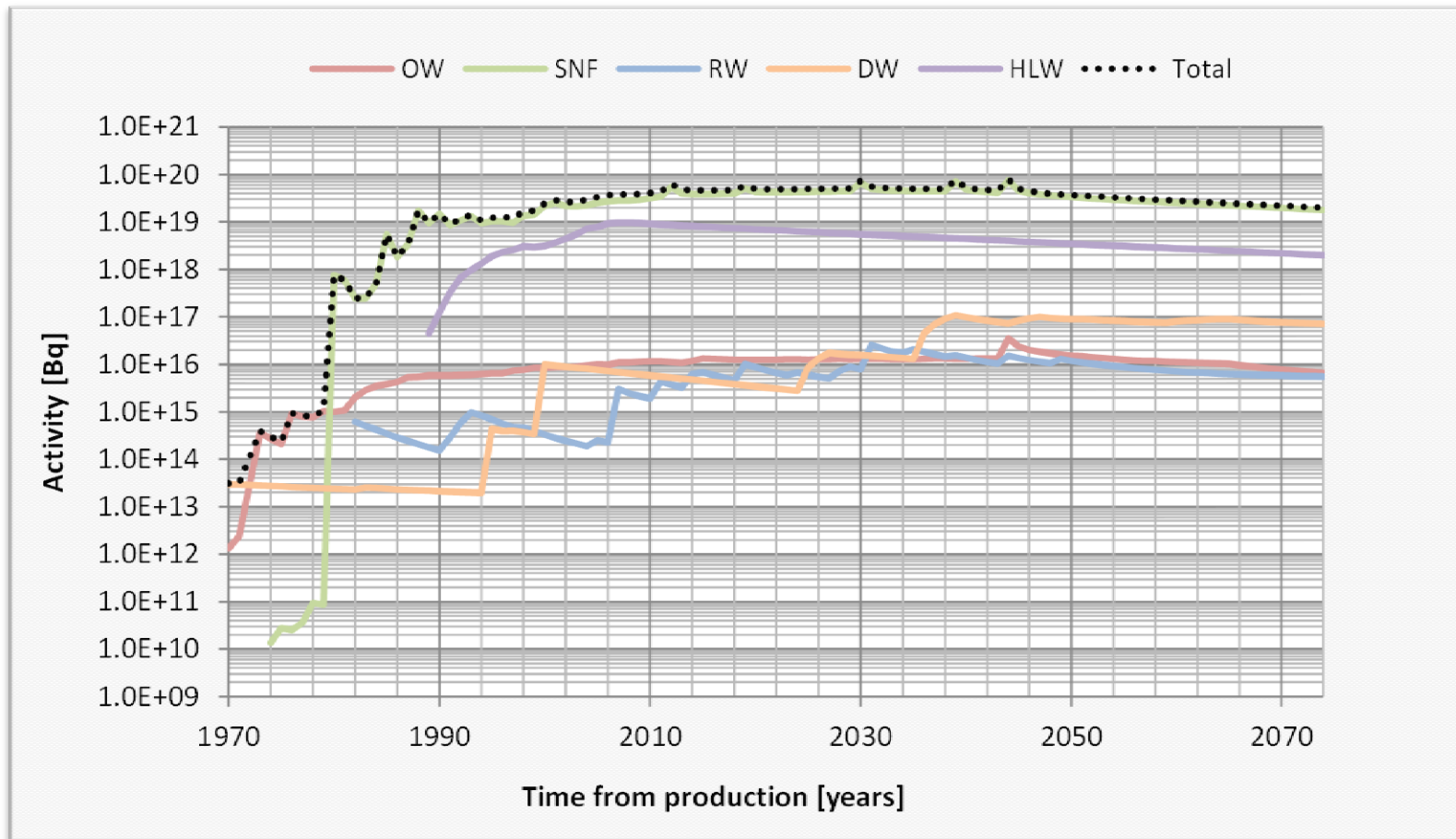
- **Goal:** Determination of radionuclide inventory for long term safety assessment in geological repository.
- 1. Definition/development of representative FAs categories, according to power plant, fuel type and fuel characteristic (BU, enrichment).

AGT	NPP	Type	²³⁵ U/Pu [wt%]	BU [Gwd/t]	Cask
J-X-950001	KKW	UO ₂	3.5	40	BE-X
.....	KKW	UO ₂	3.7	45	BE-X
J-X-950501	KKW	MOX	0.71/2.7	30	BE-Y
.....	KKW	MOX	0.71/2.6	40	BE-Y
J-Y-950001	KKW	UO ₂	3.5	48	BE-X
.....	KKW	UO ₂	4.0	50	BE-X
.....					

- 2. Development of corresponding code specific libraries (fuel type and NPP).
- 3. Calculation of radionuclides inventory for all spent fuel.
- 4. The corresponding calculated radionuclides inventory is imported in a database (MIRAM), according to each defined categories. MIRAM is the Nagra **Model Inventory for Radioactive waste Management**.

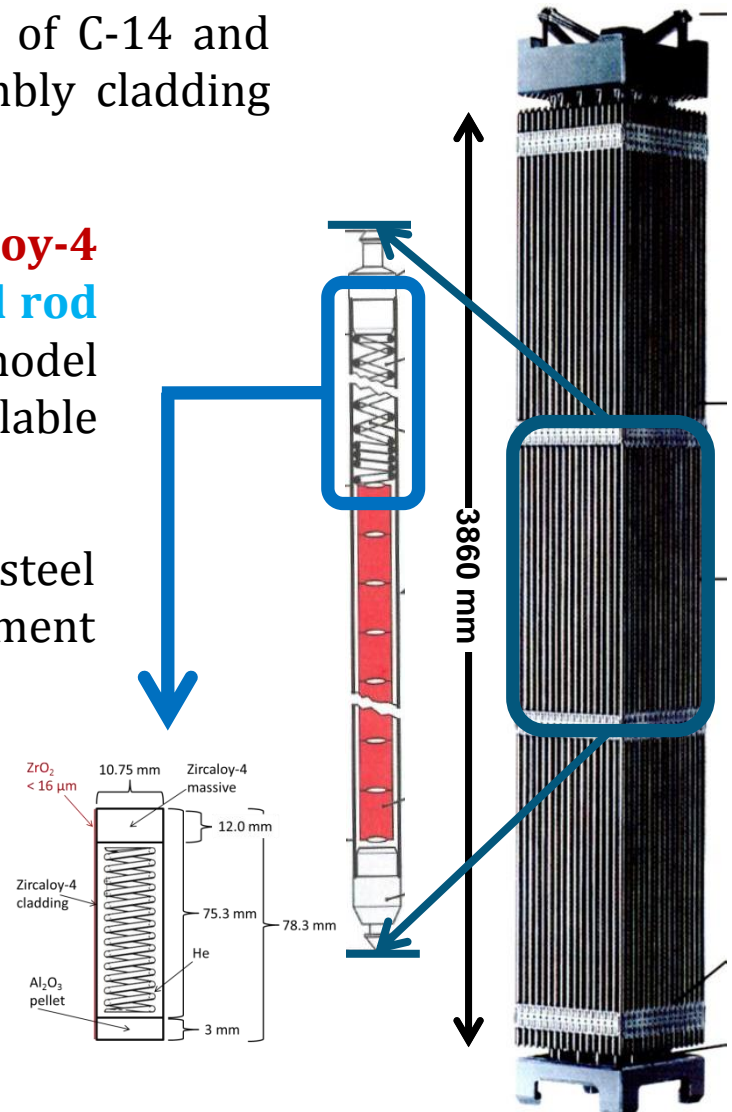
IV. Application: Radionuclide inventory

- Total waste activity build-up, according to the main MIRAM categories.



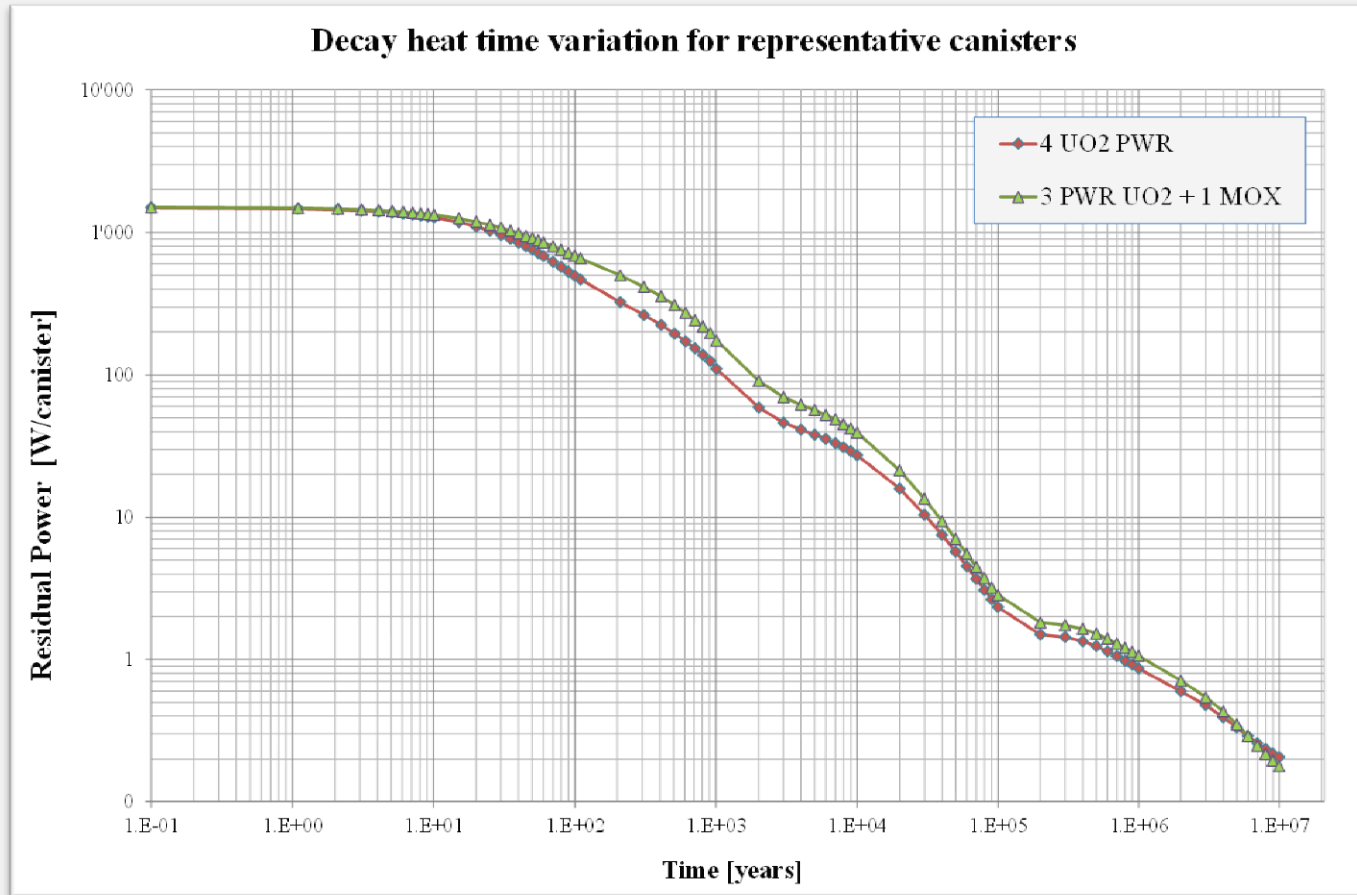
IV. Activation products calculations

- **Example** from the **CAST** project: Estimation of C-14 and activation products in LWR spent fuel assembly cladding and structural materials.
- **Task**: Determining inventory of ^{14}C in **Zircaloy-4** and **stainless steel** of an UO_2 PWR **spent fuel rod** segment with a fuel depletion/activation model (Nagra) and comparison against available experimental data from **KIT**.
- **Samples**: Zircaloy-4 cladding and stainless steel plenum spring sampled from a fuel rod segment irradiated in the Swiss Gösgen PWR
 - Average burn-up: 50.4 GWd/ t_{HM}
 - Irradiation duration: 1226 days
- **Method**:
 - Material irradiated using ORIGEN-S
 - Employing the **cladding libraries** produced with TRITON



IV. Application: Decay heat in disposal canister

- **Goal:** Determination of decay heat as function of time for long term safety assessment in geological repository.

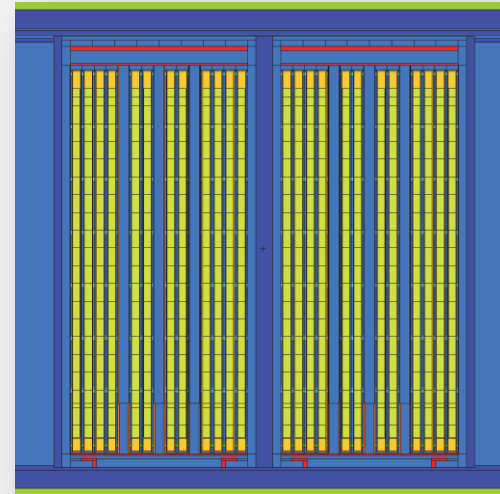
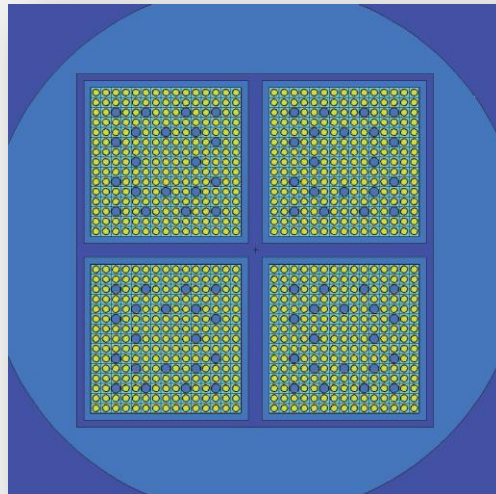


IV. Application: Gamma, neutron source terms

- Determination of neutron and gamma source terms to be input for **shielding design** and **operational safety** assessment.
- The model validation has found relevant applications in the framework of the collaboration between nuclear utilities and Nagra, in relation to the source term evaluation and related aspects:
 - An Extension of the Validation of isotopic predictions for MOX Spent Fuel, using MALIBU' and PROTEUS program in the framework of transport/storage cask licensing
 - Qualification of ORIGEN-ARP isotopic predictions for BWR Spent Fuel Assemblies in the framework of transport/storage cask licensing. This recent work is already approved by the regulator.
 - *M. Gutierrez "Development and validation of 2D and 3D SCALE LWR fuel assembly models for burnup calculations and activation studies", M.Sc. Thesis in Nuclear Engineering ETHZ (2015)*

IV. Application: Burup Credit for criticality safety

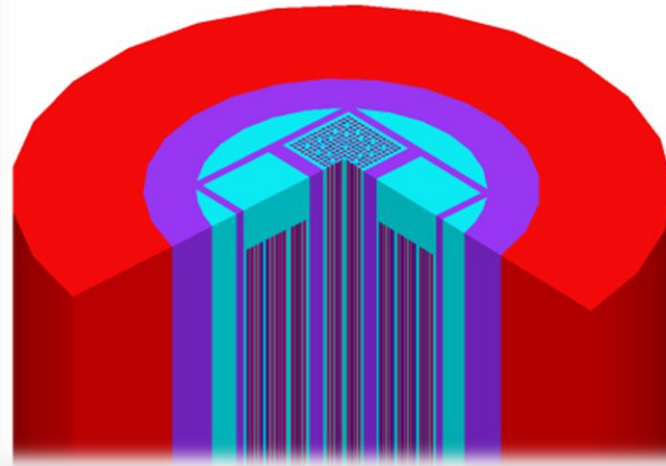
- ✓ **Goal:** Assessment of the Criticality Safety for 1 Million year, though the employment of Burnup Credit
- ✓ Cooperation Nagra / PSI: BUCSS-R Project
- ✓ The evolution of K-eff for different discharge burnup values was studied for both “actinides only” and “actinides + fission products” cases.
- ✓ Example of flooded canister, loaded with four UO₂ (**MCNPX**)



Reference: J. J. Herrero, M. Pecchia, H. Ferroukhi, S. Canepa, A. Vasiliev, S. Caruso, “Computational scheme for burnup credit applied to long term waste disposal”, International Conference on Nuclear Criticality Safety, Charlotte, NC, September 13-17, 2015

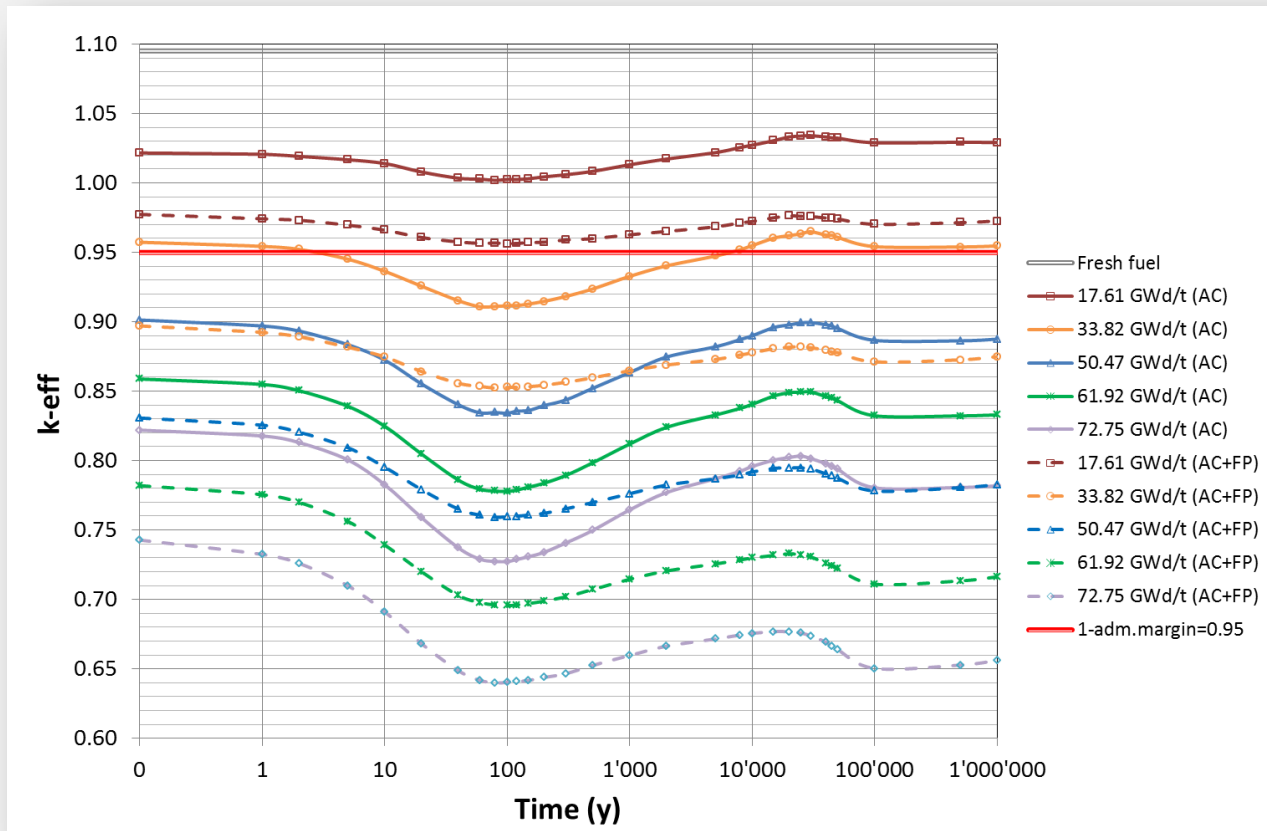
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- ✓ Example of flooded canister, loaded with four UO₂ (**KENO-VI**)



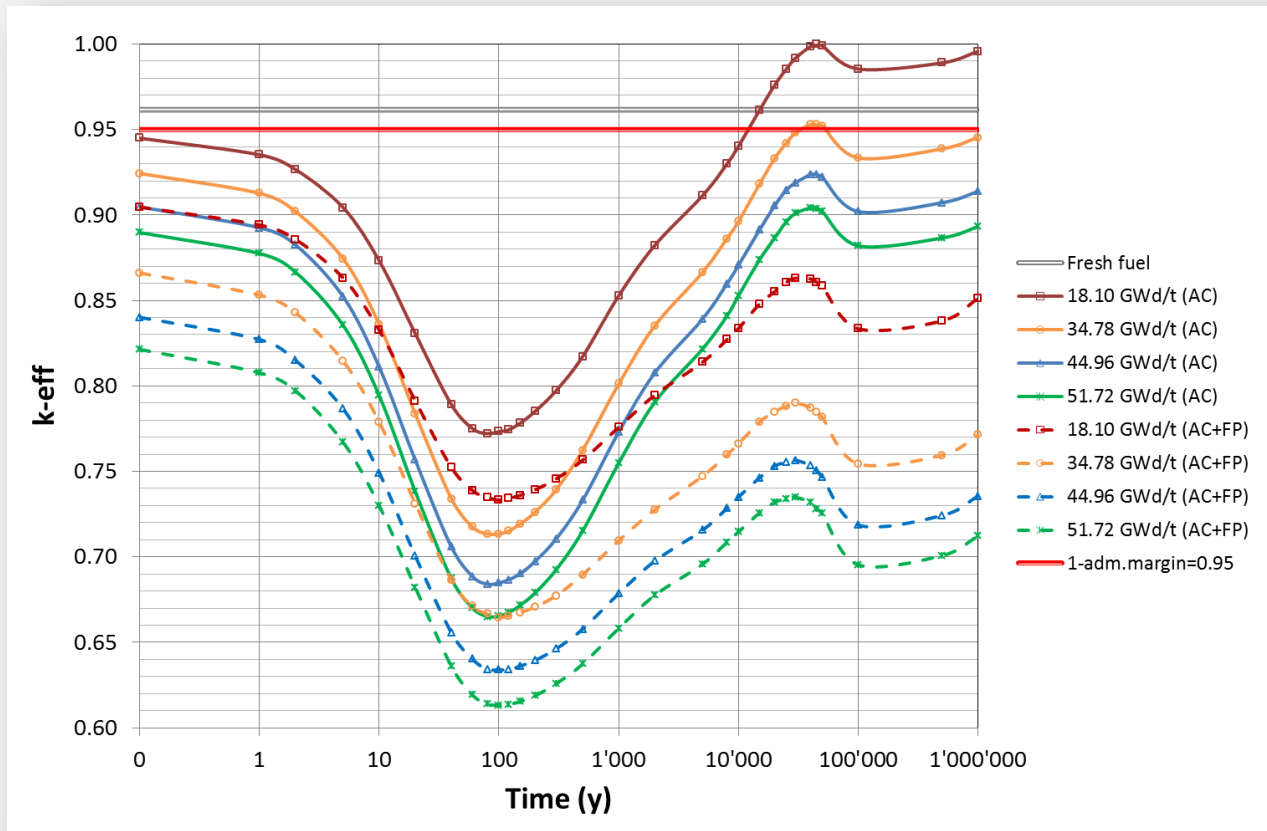
IV. Criticality Safety Assessment: BU-Credit

Evolution of k_{eff} for the intact canister loaded with UO_2 fuel, at 5 wt% ^{235}U



IV. Criticality Safety Assessment: BU-Credit

Evolution of k_{eff} for the intact canister loaded with MOX fuel



Conclusions & Further steps

- The methodology proves to be fast and reliable, being based on 2-steps:
 - Build-up of **SCALE/Triton** libraries
 - ✓ with large computational time, but needed only in the first phase
 - **Origen-Arp** and **Origen-S** standalone depletion and activation calculations
 - ✓ with short computational time and large flexibility

Further work is still needed:

- Further development and validation of other fuel assembly models
- 3D depletion/activation models to address properly the axial heterogeneities
- Test of new reactor physics codes



thank you
for your attention
nagra ●