KNS 2022 Autumn Conference

Selection of Reference PWR Spent Nuclear Fuel and Source Term Calculation for Deep Geological Disposal System

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Computational Transport & Reactor Physics Laboratory

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1. Introduction

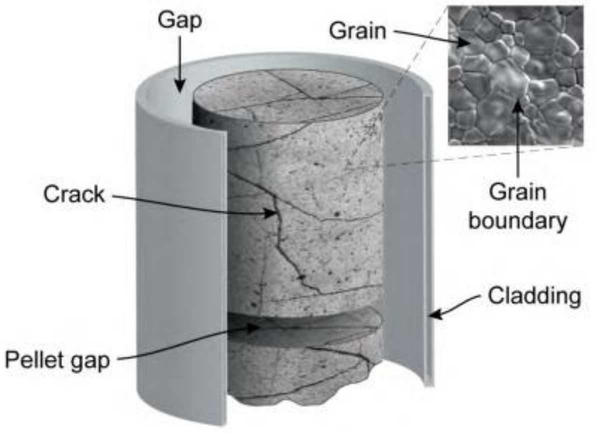

Fig. 1. Illustration of a section of a spent nuclear fuel rod, showing cracked fuel pellet with grain boundaries and gaps [1)]

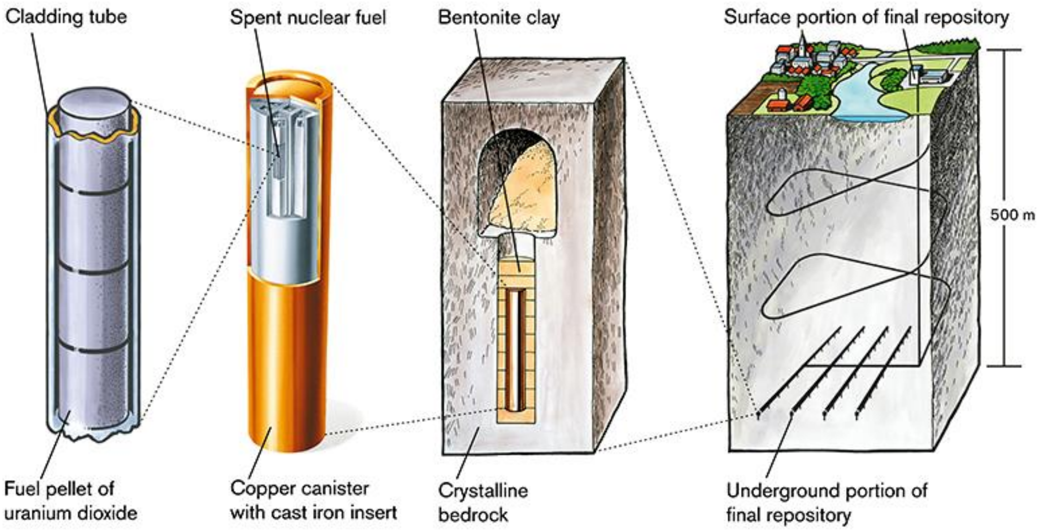

Fig. 2. The KBS-3 system [2)]

- To evaluate the safety of deep geological disposal, it is necessary;
  - to establish a design basis reference SNFs that considers the characteristics of the SNFs generated in domestic nuclear power plants,
  - to evaluate the source term of design basis reference SNFs for disposal canister safety (decay heat, criticality, radiation shielding, etc.), and
  - to quantitatively evaluate the leakage or release characteristics of nuclide isotopes in SNFs for long-term safety evaluation.
- It was possible to refer to the Korea PWR SNFs database (up to December 2021) from Korea Hydro & Nuclear Power (KHNP) through the Korea Radioactive Waste Agency (KORAD).
- Using the SCALE 6.2.4/TRITON and ORIGEN-ARP computer code, the source term evaluation was performed on the design basis reference SNF.

1) POSIVA 2013-01, Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto, 2013
2) SKB TR-99-06, Deep repository for spent nuclear fuel SR 97 – Post-closure safety, 1999

2. Design Basis Reference SNFs (1/5)

- Databases of the 20,970 PWR SNFs generated from 1979 to 2021 was provided by KHNP.
- For each assembly, the database contains:
  - initial uranium mass (grams),
  - initial enrichment (wt% ^{235}U),
  - discharge burnup (MWd/MTU),
  - specific power (MW/MTU),
  - effective full power days (EFPDs),
  - discharged date (dd/mm/yyyy),
  - assembly lattice type.

- The initial enrichment is the value of the normal pins.
- The initial uranium mass is the sum of the normal pins, zoned pins, and burnable poison pins.
- The burnup is the final burnup after the assembly has been discharged from the reactor.

2. Design Basis Reference SNFs (2/5)

❑ Distribution of SNFs properties (1/3)

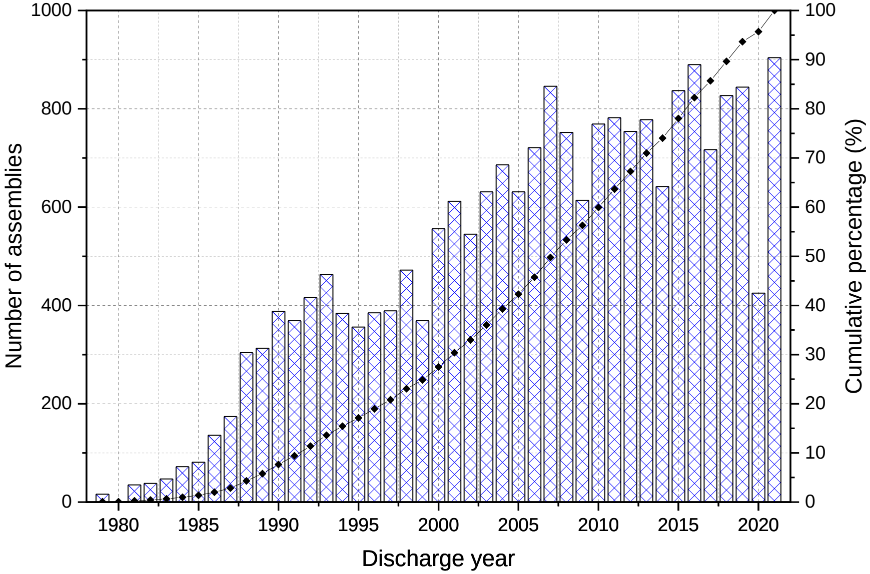

Fig. 3. Annual number of fuel assemblies discharged

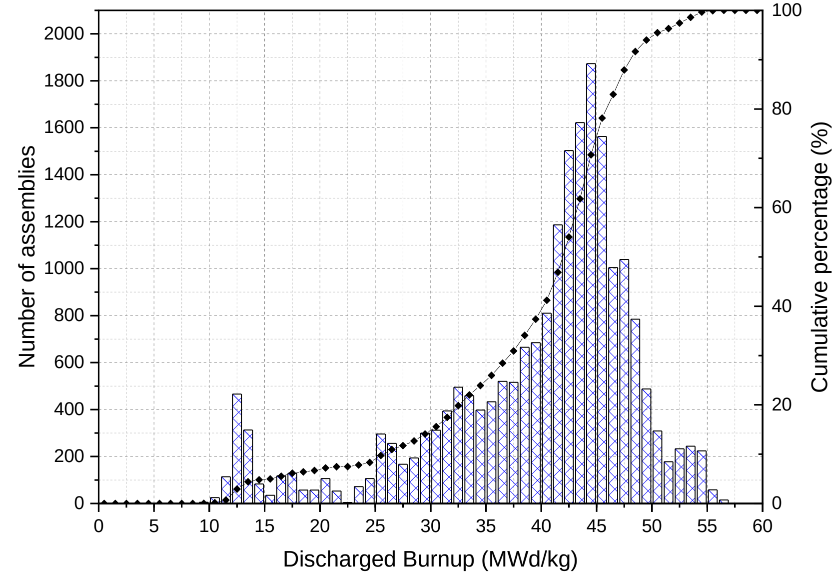

Fig. 4. Number of assemblies as a function of discharged burnup (x-axis bin: 1 MWd/kg)

Table I. Number of SNFs by burnup range

Burnup range (MWd/kg)	Number of SNFs	Percentage (%)
Over 5 – 15 or less	1,007	4.80
Over 15 – 25 or less	737	3.51
Over 25 – 35 or less	3,271	15.60
Over 35 – 45 or less	9,811	46.76
Over 45 – 55 or less	6,071	28.95
Over 55 – 65 or less	73	0.35

- Average burnup: 40 MWd/kg
- Median burnup: 42 MWd/kg

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2. Design Basis Reference SNFs (3/5)

Distribution of SNFs properties (2/3)

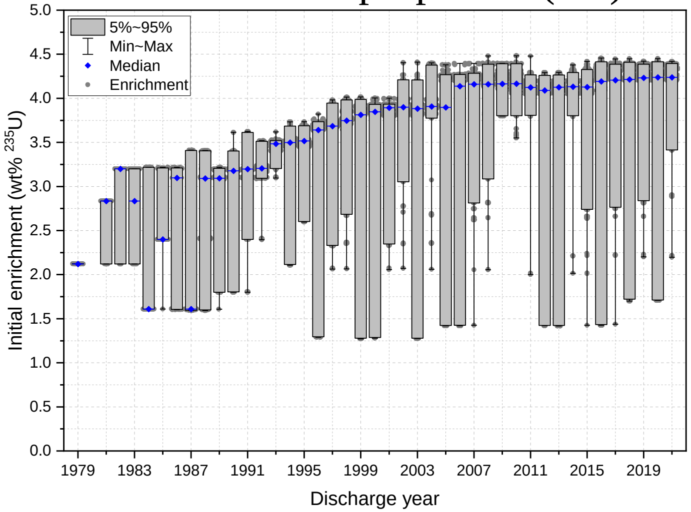

Fig. 5. Evolution of the initial enrichment as a function of fuel discharge year for SNFs

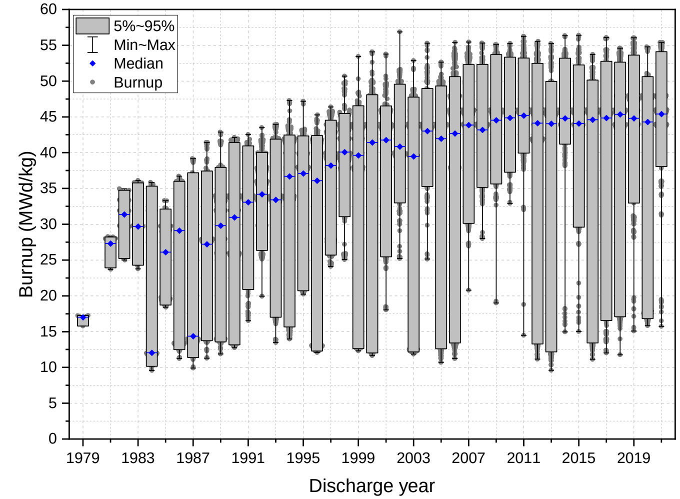

Fig. 6. Evolution of the burnup as a function of fuel discharge year for SNFs

Table II. Enrichment and burnup by discharge year group

	Discharge year	Initial enrichment (wt% ^{235}U)					Burnup (MWd/MTU)					# of SNFs
		Min.	Median	Max.	Under 5%	Upper 5%	Min.	Median	Max.	Under 5%	Upper 5%	20,969
PWR-1	1979-1996	[1.29	3.20	3.74]	[1.60	3.66]	[9,563	32,458	47,279]	[12,961	41,380]	3,921 (19%)
PWR-2	1997-2008	[1.27	3.89	4.41]	[2.31	4.27]	[10,697	41,908	56,859]	[23,892	49,511]	7,203 (34%)
PWR-3	2009-2021	[1.42	4.17	4.48]	[2.64	4.38]	[9,609	44,676	56,385]	[18,500	52,885]	9,845 (47%)

- Burnup and initial enrichment have a correlation.
- The median enrichment and burnup in general increases over time.
- The development of second-generation fuels (e.g., PLUS7, ACE7) with the increased allowed burnup.
  - Allowable burnup of PLUS7: ≤ 55 MWd/kg
  - Acceptable burnup of ACE7: ≥ 55 MWd/kg

2. Design Basis Reference SNFs (4/5)

❑ Distribution of SNFs properties (3/3)

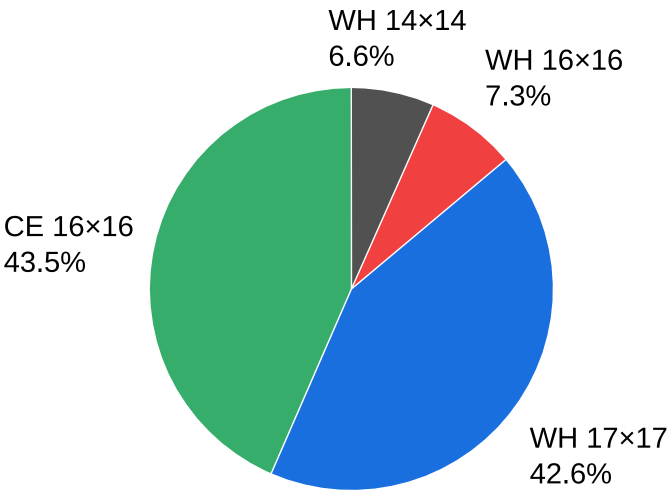

Fig. 7. Portions of assembly lattice types in SNF database

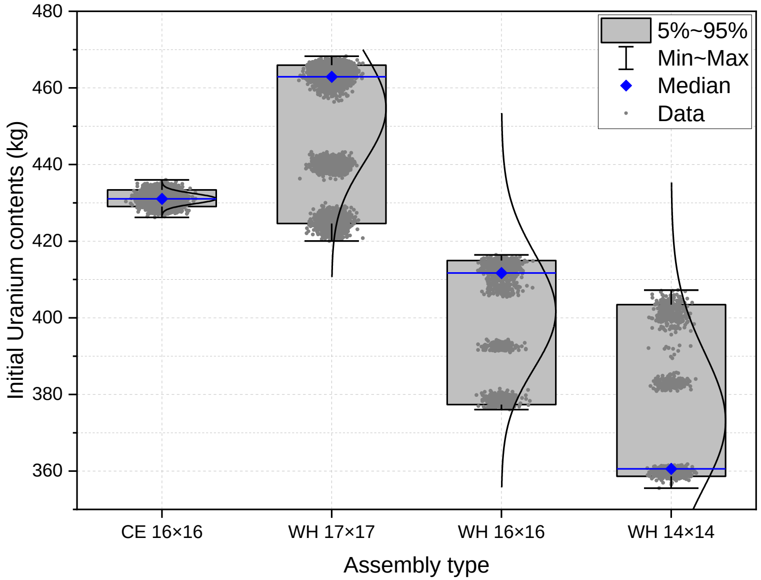

Fig. 8. Initial uranium mass by assembly type

- WH 17×17 and CE 16×16 type assemblies compose ~86% of the total SNFs.
- The maximum initial uranium mass of WH 17×17 and CE 16×16 are 468 kg and 435 kg, respectively.
- The initial uranium mass and burnup are important parameters for evaluating radioactivity, decay heat, criticality, radiation shielding safety, and inventory of SNFs.

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2. Design Basis Reference SNFs (5/5)

- Selection of design basis reference SNFs
  - A scenario was assumed in which 70 % of the total capacity of the deep geological disposal system was to be disposed of low burnup SNFs, and the remaining 30 % of the SNFs with high burnup.

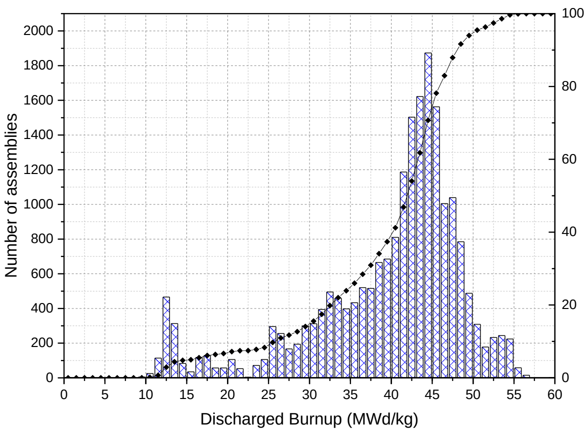


Fig. 9. Number of assemblies as a function of discharged burnup
(x-axis bin: 1 MWd/kg)

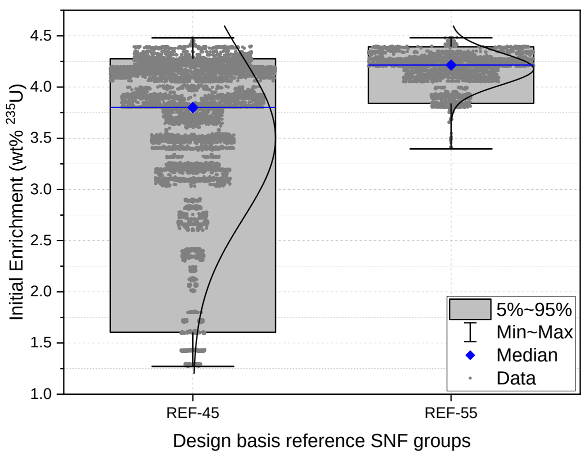


Fig. 10. Distribution of initial enrichment for design basis reference SNF groups

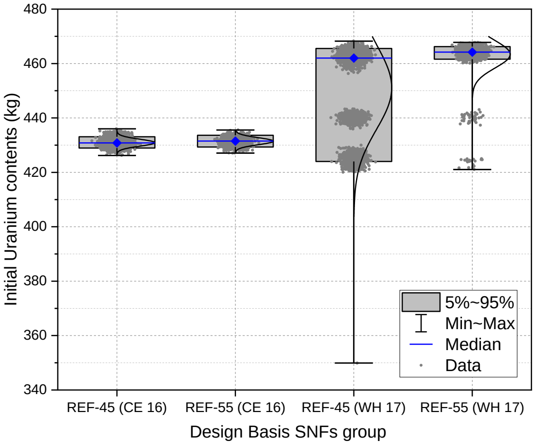


Fig.11. Distribution of initial uranium mass for design basis reference SNF groups

Table III. Summary of design basis reference SNFs

Design basis reference SNFs	Lattice type	Burnup [MWd/MTU]	Enrichment [wt% U^{235}]	Uranium mass [kg]	Specific power [MW/MTU]	Irradiation time [EFPDs]
REF-PLUS7-LU	16×16	45,000	4.5	436	40	1,125
REF-PLUS7-HU	16×16	55,000	4.5	436	40	1,375
REF-ACE7-LU	17×17	45,000	4.5	468	40	1,125
REF-ACE7-HU	17×17	55,000	4.5	468	40	1,375

3. Modeling and Computational Code (1/2)

- Fuel assembly modeling

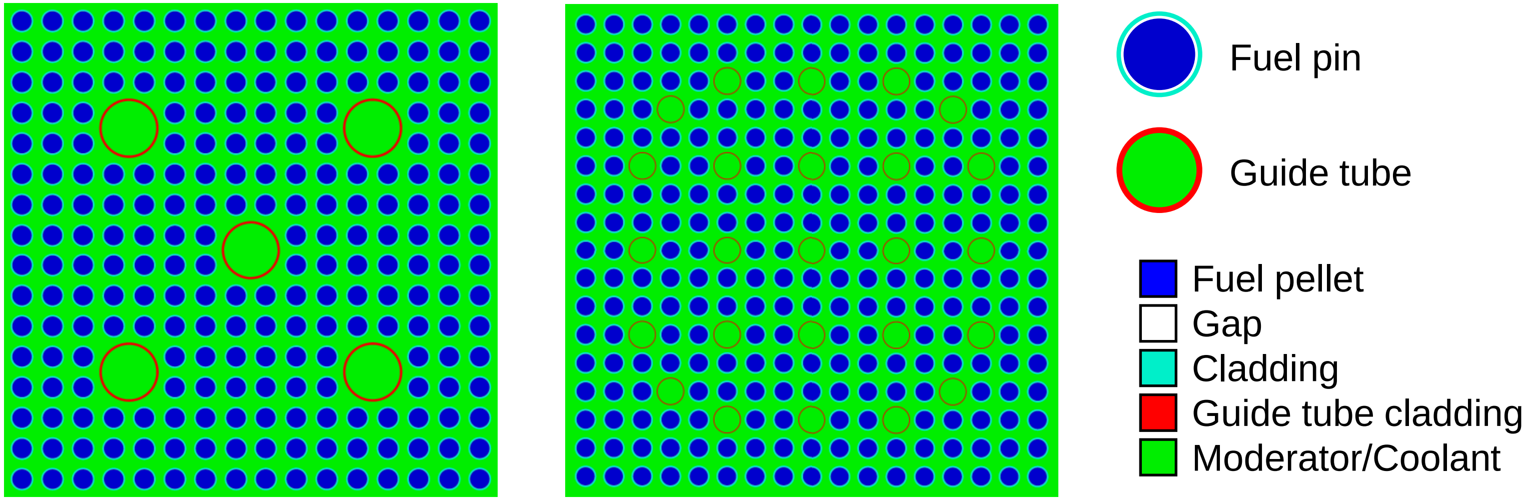


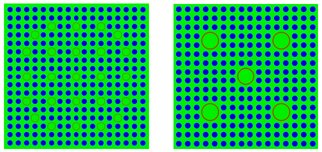
Fig. 12. Configuration of fuel assembly (left: PLUS7, right: ACE7)

Table IV. PLUS7 and ACE7 design parameters

Design parameter	PLUS7	ACE7
Fuel rod array	16×16	17×17
Number of fuel rods	236	264
Active fuel length	381	365.76
Number of guide tube	4	24
Number of instrumentation tube	1	1
Fuel assembly length [cm]	452.8	406.3
Fuel assembly pitch [cm]	20.7772	21.5040
Fuel rod length [cm]	409.4	388.1
Pin pitch [cm]	1.285	1.260
Fuel diameter [cm]	0.8192	0.8192
Cladding material	ZIRLO	ZIRLO
Cladding I.D. [cm]	0.8357	0.8357
Cladding O.D. [cm]	0.95	0.95
Guide tube material	ZIRLO	ZIRLO
Guide tube I.D. [cm]	2.286	1.008
Guide tube O.D. [cm]	2.4892	1.224

3. Modeling and Computational Code (2/2)

❑ SCALE 6.2.4/TRITON and ORIGEN-ARP



TRITON

Calculate a 2-D deterministic transport.

Libraries

Pregenerate the ORIGEN cross section libraries using TRITON for different
- fuel types,
- initial enrichments,
- moderator densities,

with full details of fuel design and operating conditions.

Interpolate the libraries based on
- fuel type,
- specific burnup,
- enrichment,
- moderator density.

Provide
- cycle history,
- decay time,
- initial fuel composition.

ORIGEN-ARP

Perform depletion and decay calculations according to specific conditions and using customized libraries.

Nuclide compositions.

Neutron and photon source terms.

Activity, decay heat, gamma heat, etc.

Fig. 13. The TRITON/ORIGEN procedure used in this work to generate the design basis reference SNFs calculation [1)]

1) NUREG/CR-7227, US Commercial Spent Nuclear Fuel Assembly Characteristics: 1968-2013, 2015

4. Results (1/5)

- Activity, decay heat, gamma heat, and plutonium contents

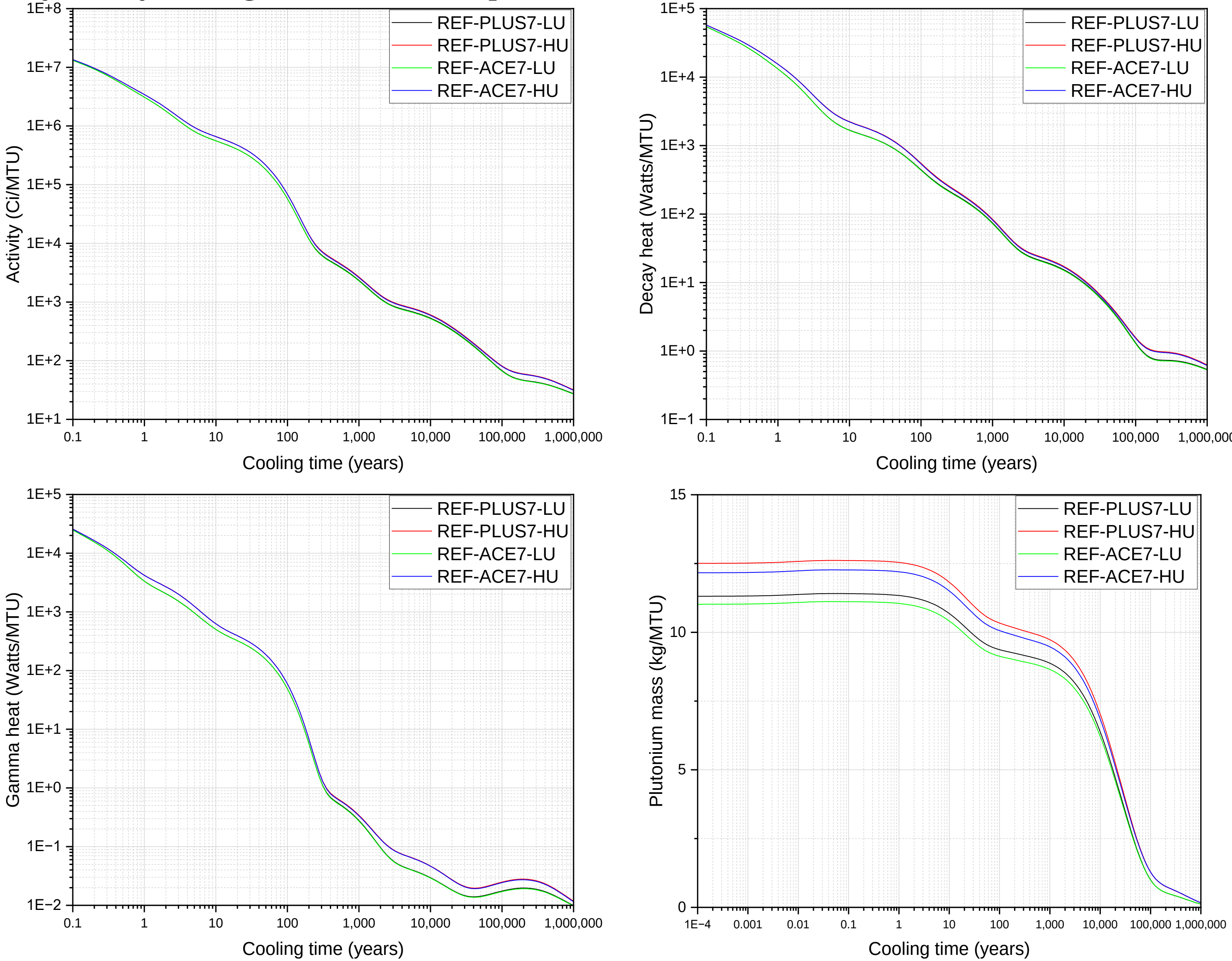


Fig. 14. (a) Activity, (b) Decay heat, (c) Gamma heat, and (d) Pu mass as a function of cooling time for each design basis reference SNFs

4. Results (2/5)

- Estimation of cooling time to satisfy the decay heat criteria in the disposal canister (1/2)

Sweden KBS-3 Disposal System (TR-10-13) - Requirements related to repository design and long-term safety
Requirement on handling: *The fuel assemblies to be encapsulated in any single canister shall be selected with respect to burnup and age so that the total decay power in the canister will not result in temperatures exceeding the maximum allowed in the buffer.*
Criterion: *The total decay power in* ***each canister must not exceed 1,700 W***

POSIVA 2005-02 – Disposal Canister for Spent Nuclear Fuel – Design Report
The maximum canister decay power is limited in reference case to 1,700 W for the BWR canister in safety analysis. This leads to comparable power of 1,370 W for VVER 440 canister and of ***1,830 W power for EPR canister***

Fig. 15. Left: Canister geometries, Right: Canister overpack is made out of copper and the insert is cast iron [1)]

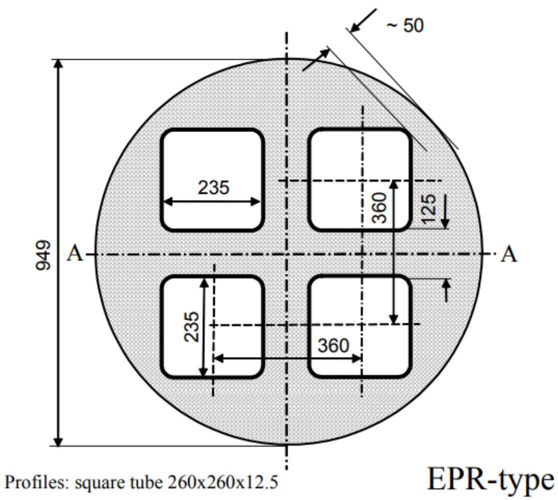


Fig. 16. The section of the insert for EPR fuel assemblies [2)]

1) POSIVA 2013-01, Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto, 2013
2) POSIVA 2005-02, Disposal Canister for Spent Nuclear Fuel – Design Report, 2005

4. Results (3/5)

- Estimation of cooling time to satisfy the decay heat criteria in the disposal canister (2/2)

Table V. Uranium mass per canister

Design basis reference SNFs	Median uranium mass (grams)	Canister uranium mass (MTU)
REF-PLUS7-LU	430,834	1.72334
REF-PLUS7-HU	431,505	1.72602
REF-ACE7-LU	461,992	1.84797
REF-ACE7-HU	464,196	1.85678

Table VI. Decay heat per canister as a function of cooling time

Cooling time (years)	REF-PLUS7-LU	REF-PLUS7-HU	REF-ACE7-LU	REF-ACE7-HU
3	7,703	9,765	8,210	10,432
5	4,559	6,021	4,853	6,422
10	2,862	3,820	3,044	4,068
20	2,198	2,885	2,336	3,070
30	1,866	2,422	1,981	2,574
40	1,626	2,093	1,725	2,222
50	1,380	1,760	1,461	1,867
56	1,258	1,598	1,331	1,693
64	1,140	1,441	1,205	1,525
72	1,027	1,292	1,084	1,366
81	920	1,153	971	1,218
92	823	1,026	867	1,082
104	734	912	773	960

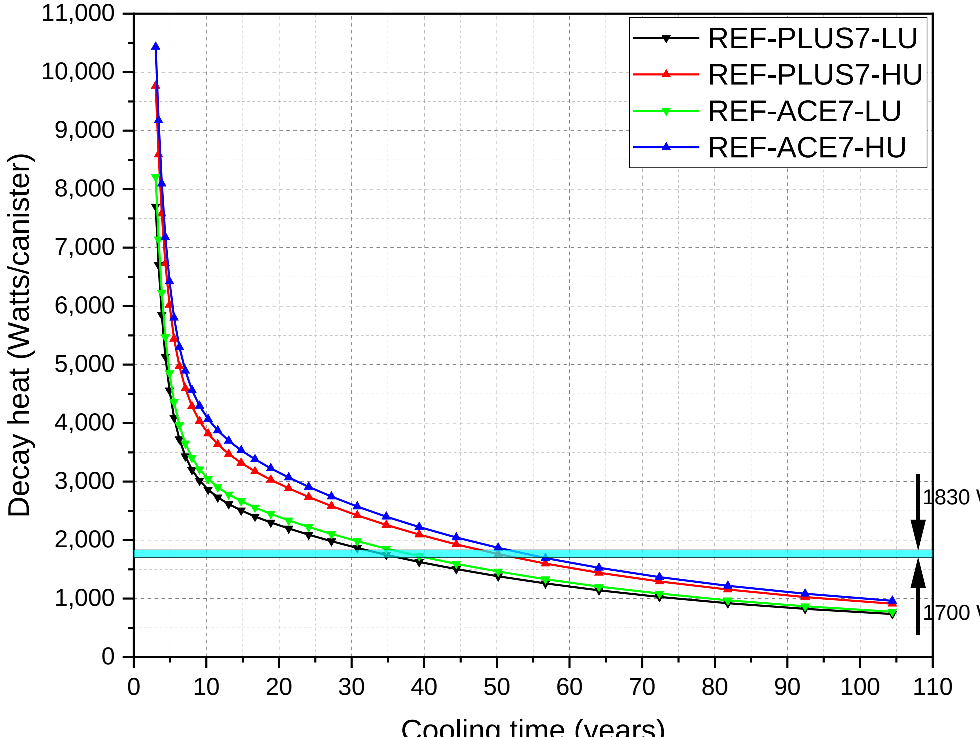


Fig. 17. Decay heat per canister as a function of cooling time

4. Results (4/5)

- Phases D) The selection of the most safety-relevant radionuclides for the release, retention, and transport calculations (41 radionuclides were finally screened).

STUK's regulatory guidelines YVL D.5 (STUCK 2011a, paragraph 313). The radionuclides mentioned in YVL D.5 are the following:

- *Long-lived alpha-emitting Ra, Th, Pa, Pu, Am, and Cm isotopes,*
- *Long-lived uranium isotopes,*
- *C-14, Cl-36, Ni-59, Se-79, Zr-93, Nb-94, Tc-99, Pd-107, I-129, Sn-126, Cs-135, Np-237*

Ag-108m,	Am-241,	Am-243,	Be-10,	C-14,	Cl-36,	Cm-245,	Cm-246,	Cs-135,	Cs-137,
I-129,	Mo-93,	Nb-91,	Nb-92,	Nb-93m,	Nb-94,	Ni-59,	Ni-63,	Np-237,	Pa-231,
Pd-107,	Pu-238,	Pu-239,	Pu-240,	Pu-241,	Pu-242,	Se-79,	Sm-151,	Sn-126,	Sr-90,
Ra-226,	Tc-99,	Th-229,	Th-230,	Th-232,	U-233,	U-234,	U-235,	U-236,	Pu-238,
Zr-93.									

*Blue nuclides: The instant release fractions (IRF)

Table VII. Fuel parameters are taken into account in the reference inventory (based on Anttila 2005, Appendix 2)

Parameter	Values in Anttila (2005)
Initial enrichment (wt%)	3.8, 4.2
Discharge burnup (MWd/kg)	45, 50, 60

* POSIVA 2013-01, Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto, Section 7.3, 2013
* KAERI/AR-1325/2021, 처분환경에서 사용후핵연료 중 장반감기 방사성 핵종의 순간 누출 분율에 관한 기술현황 분석, 2021
* SKB 2010f, Data report for the safety assessment SR-Site, TR-10-52
* Nykyri, et. al., Experimental and modeling investigations of the biogeochemistry of gas production from low and intermediate level radioactive waste, Applied Geochemistry, Vol. 23, no. 6, p. 1383-1418

4. Results (5/5)

Table VIII. The inventory (GBq/MTU) of the potentially safety-relevant radionuclides in one metric ton uranium (MTU) as a function of cooling time

Radionuclide	Reference 30 years cooling	30 years cooling	40 years cooling	44 years cooling	50 years cooling	56 years cooling	64 years cooling	72 years cooling
Ag-108m	2.50E+04	6.61E-02	6.52.E-02	6.47.E-02	6.41.E-02	6.34.E-02	6.27.E-02	6.19.E-02
Am-241	1.93E+05	2.10E+05	2.27.E+05	2.34.E+05	2.40.E+05	2.44.E+05	2.46.E+05	2.47.E+05
Am-243	3.42E+03	2.24E+03	2.24.E+03	2.24.E+03	2.24.E+03	2.24.E+03	2.24.E+03	2.24.E+03
Be-10	1.26E-02	9.92E-05	9.92.E-05	9.92.E-05	9.92.E-05	9.92.E-05	9.92.E-05	9.92.E-05
C-14	1.61E+02	5.54E+00	5.53.E+00	5.53.E+00	5.52.E+00	5.52.E+00	5.52.E+00	5.51.E+00
Cl-36	2.63E+00	1.64E-73	1.64.E-73	1.64.E-73	1.64.E-73	1.64.E-73	1.64.E-73	1.64.E-73
Cm-245	1.03E+02	8.75E+01	8.74.E+01	8.74.E+01	8.73.E+01	8.73.E+01	8.72.E+01	8.72.E+01
Cm-246	3.57E+01	2.17E+01	2.17.E+01	2.17.E+01	2.17.E+01	2.17.E+01	2.16.E+01	2.16.E+01
Cs-135	3.43E+01	2.85E+01	2.85.E+01	2.85.E+01	2.85.E+01	2.85.E+01	2.85.E+01	2.85.E+01
Cs-137	3.46E+06	3.16E+06	2.60.E+06	2.31.E+06	2.02.E+06	1.74.E+06	1.47.E+06	1.21.E+06
I-129	1.91E+00	1.68E+00	1.68.E+00	1.68.E+00	1.68.E+00	1.68.E+00	1.68.E+00	1.68.E+00
Mo-93	2.26E+01	1.74E-06	1.73.E-06	1.73.E-06	1.73.E-06	1.73.E-06	1.73.E-06	1.72.E-06
Nb-91	2.86E-04	1.18E-06	1.17.E-06	1.17.E-06	1.16.E-06	1.15.E-06	1.14.E-06	1.13.E-06
Nb-92	2.35E-04	2.05E-09	2.05.E-09	2.05.E-09	2.05.E-09	2.05.E-09	2.05.E-09	2.05.E-09
Nb-93m	5.08E+03	7.82E+01	8.59.E+01	8.94.E+01	9.25.E+01	9.52.E+01	9.74.E+01	9.92.E+01
Nb-94	7.52E+02	1.60E-02	1.60.E-02	1.60.E-02	1.60.E-02	1.60.E-02	1.60.E-02	1.60.E-02
Ni-59	2.21E+02	1.84E-167	1.84.E-167	1.84.E-167	1.84.E-167	1.84.E-167	1.84.E-167	1.84.E-167
Ni-63	2.47E+04	2.33E-165	2.19.E-165	2.12.E-165	2.04.E-165	1.95.E-165	1.85.E-165	1.75.E-165
Np-237	2.37E+01	2.31E+01	2.37.E+01	2.41.E+01	2.45.E+01	2.50.E+01	2.56.E+01	2.63.E+01
Pa-231	1.39E-03	4.36E-04	5.44.E-04	6.09.E-04	6.82.E-04	7.65.E-04	8.59.E-04	9.64.E-04
Pd-107	9.72E+00	7.43E+00	7.43.E+00	7.43.E+00	7.43.E+00	7.43.E+00	7.43.E+00	7.43.E+00
Pu-238	2.64E+05	2.42E+05	2.27.E+05	2.18.E+05	2.08.E+05	1.98.E+05	1.86.E+05	1.75.E+05
Pu-239	1.42E+04	1.47E+04	1.47.E+04	1.47.E+04	1.47.E+04	1.47.E+04	1.47.E+04	1.47.E+04
Pu-240	3.12E+04	1.96E+04	1.97.E+04	1.97.E+04	1.98.E+04	1.98.E+04	1.98.E+04	1.98.E+04
Pu-241	1.75E+06	1.83E+06	1.21.E+06	9.46.E+05	7.15.E+05	5.21.E+05	3.65.E+05	2.43.E+05
Pu-242	2.17E+02	1.57E+02	1.57.E+02	1.57.E+02	1.57.E+02	1.57.E+02	1.57.E+02	1.57.E+02
Ra-226		1.83E-05	3.63.E-05	5.12.E-05	7.23.E-05	1.02.E-04	1.44.E-04	2.03.E-04
Se-79	4.67E+00	4.42E+00	4.42.E+00	4.42.E+00	4.42.E+00	4.42.E+00	4.42.E+00	4.42.E+00
Sm-151	1.74E+04	1.36E+04	1.28.E+04	1.23.E+04	1.17.E+04	1.12.E+04	1.06.E+04	9.90.E+03
Sn-126	3.92E+01	1.56E+01	1.56.E+01	1.56.E+01	1.56.E+01	1.56.E+01	1.56.E+01	1.56.E+01
Sr-90	2.23E+06	2.06E+06	1.68.E+06	1.48.E+06	1.29.E+06	1.10.E+06	9.24.E+05	7.56.E+05
Tc-99	8.48E+02	7.82E+02	7.82.E+02	7.82.E+02	7.82.E+02	7.82.E+02	7.82.E+02	7.82.E+02
Th-229		1.37E-05	1.67.E-05	1.87.E-05	2.14.E-05	2.48.E-05	2.92.E-05	3.49.E-05
Th-230	1.32E-02	3.86E-03	6.04.E-03	7.56.E-03	9.44.E-03	1.18.E-02	1.47.E-02	1.83.E-02
Th-232	2.06-E08	4.66E-08	5.26.E-08	5.63.E-08	6.04.E-08	6.50.E-08	7.03.E-08	7.62.E-08
U-233	3.86E-03	3.21E-03	4.08.E-03	4.61.E-03	5.22.E-03	5.92.E-03	6.74.E-03	7.68.E-03
U-234	5.53E+01	2.50E+01	3.07.E+01	3.39.E+01	3.73.E+01	4.11.E+01	4.51.E+01	4.93.E+01
U-235	8.15E-01	6.00E-01	6.00.E-01	6.00.E-01	6.00.E-01	6.00.E-01	6.01.E-01	6.01.E-01
U-236	1.46E+01	1.44E+01	1.44.E+01	1.44.E+01	1.44.E+01	1.44.E+01	1.45.E+01	1.45.E+01
U-238	1.17E+01	1.14E+01	1.14.E+01	1.14.E+01	1.14.E+01	1.14.E+01	1.14.E+01	1.14.E+01
Zr-93	1.27E+02	1.06E+02	1.06.E+02	1.06.E+02	1.06.E+02	1.06.E+02	1.06.E+02	1.06.E+02

5. Conclusion

- In order to utilize the deep geological disposal system safety case, the domestic PWR spent nuclear fuels database was analyzed and the design basis reference spent nuclear fuels were selected.
- Also, the source terms and characteristics of the spent nuclear fuels for the cooling time were analyzed.
- For each design basis spent nuclear fuels, the ORIGEN and TRITON codes were used to evaluate the decay heat, which is essential for the safety evaluation of the deep geological disposal system, and the neutron/gamma emission spectra required for radiation shielding analysis.
- From viewpoint of decay heat, the design basis spent nuclear fuels could be disposed of in a disposal canister after 50 years of cooling time.
- In the future, more detailed evaluations of radionuclides are required for the source terms considering the instant release and UO_2 matrix dissolution.

Thank you for listening

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