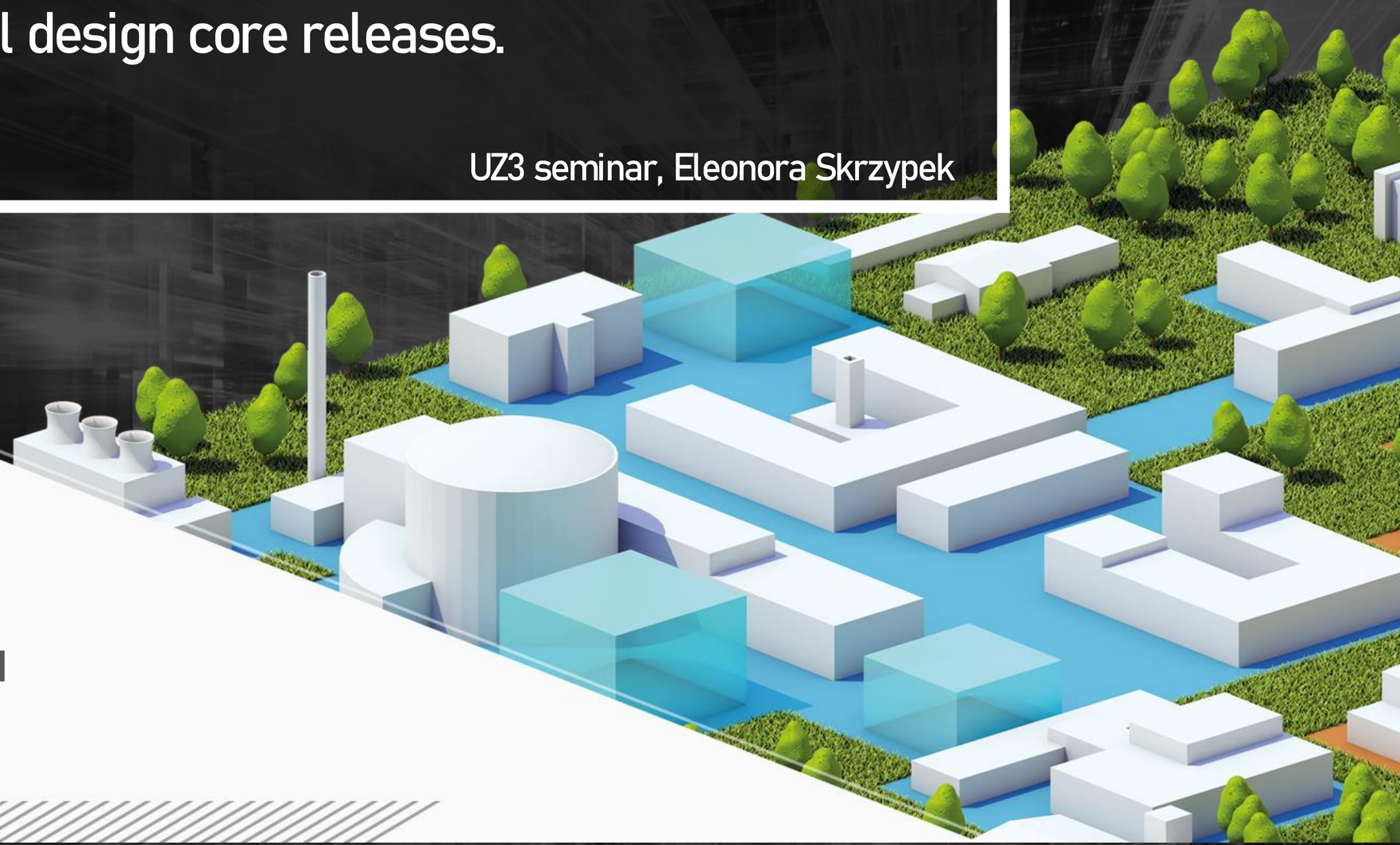
Mechanistic source term estimates simplified parametric model for the HTGR prismatic reactor for normal operation state on the example of TeResa preconceptual design core releases.

UZ3 seminar, Eleonora Skrzypek



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Abstract

Releases during normal operation – priority for the initial reactor design:

- The key issue: assurance of safety for employees and operators of the reactor, doses limitation, which they will be exposed to during normal operation.
- The determination of releases to the environment (during normal operation and emergency events) becomes an important aspect– verified by the nuclear regulatory authority (before building permit and commissioning a potential unit).
- TeResa reactor design – adapted to the adopted standards necessary to be met.

- Mechanistic source term estimates simplified parametric model for the HTGR prismatic reactor during normal operation.
- The model will be used to evaluate the core and environmental releases, based on the available data.
- The presented results of the estimation – not exceeding the prescribed standards and will guarantee social acceptance for the construction of the reactor.

TRISO characteristics

- Kernel – UO_2 or ceramic UCO.
- Tri-structural isotropic design (4 layers around the fuel kernel).
- Pressed into semi-graphitic matrix in the shape of fuel pellet (PMR – Prismatic Modular Reactor) or sphere (PBR – Pebble Bed Reactor).

Resistance:

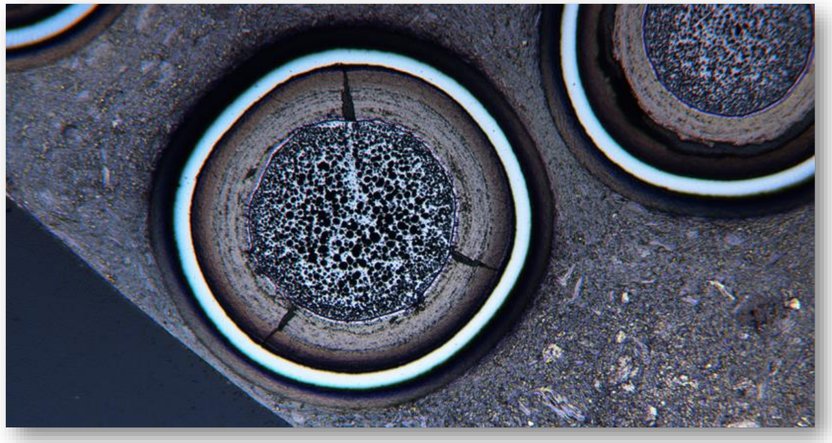
- It has been proven in many experiments (AVR), that in the event of accidents with a loss of coolant, it is possible to limit the maximum temperature to a value of 1600 ° C, and the release of radioactive substances from these fuel elements will remain very small.

Operational temperature:

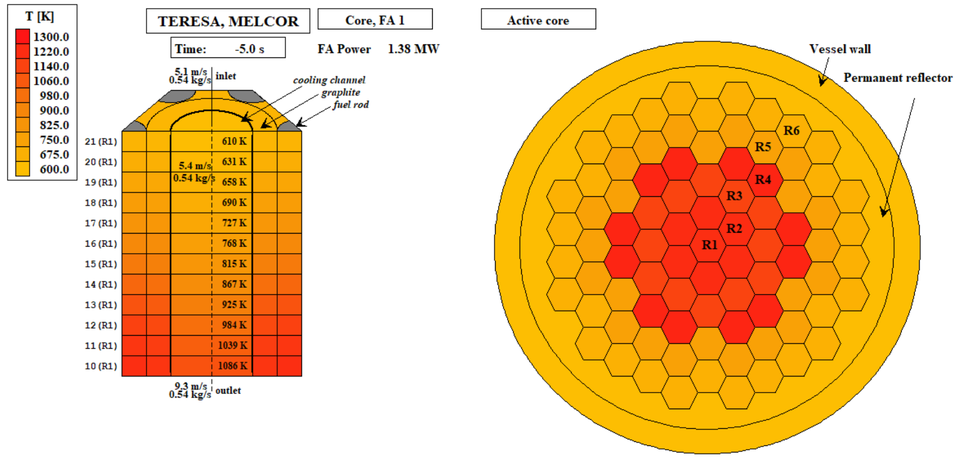
- The maximum fuel temperature during normal power plant operation will be less than 1000 ° C, and only a few percent of the particles will reach these temperatures.

Burnup:

- Achieved high burn-up values (greater than 100,000 MWd/t), high fluence of fast neutrons (up to 8 x 10^{21} n/cm^2) at very high operating temperature (T_{He} = 950°C) for experiments in AVR.



Source: https://www.energy.gov/ne/articles/triso-particles-most-robust-nuclear-fuel-earth



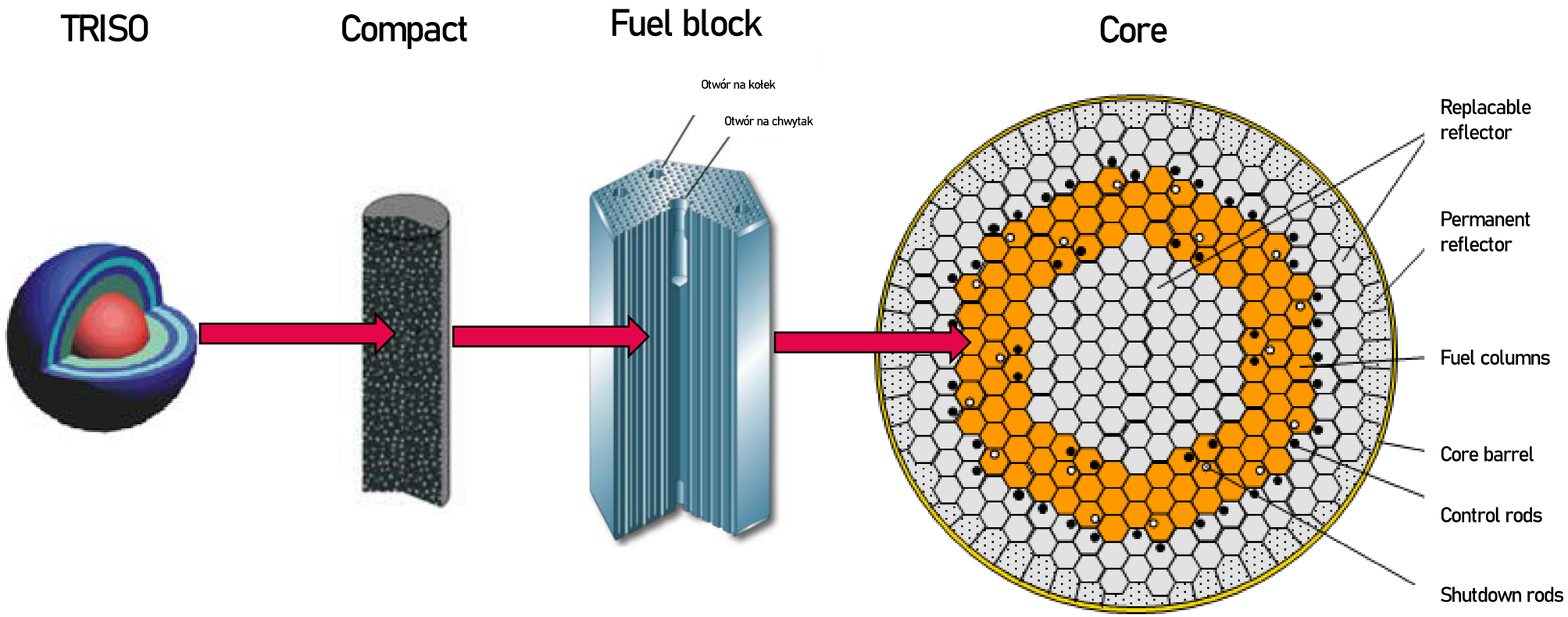
TeResa core temperatures – central column (left) and bottom active core (right)

Source: K. Kugeler and Z. Zhang, Modular High-temperature Gas-cooled Reactor Power Plant, https://doi.org/10.1007/978-3-662-57712-7_1 , Ivens G., Wimmers M., The AVR as test bed for fuel elements in AVR experimental high-temperature reactor, VDI-Verlag, Düsseldorf, 1990., Ziermann E., Ivens G., Final report on the power operation of the AVR experimental nuclear power plant, JÜL-3448, Oct. 1997.

HTGR releases mechanism summary

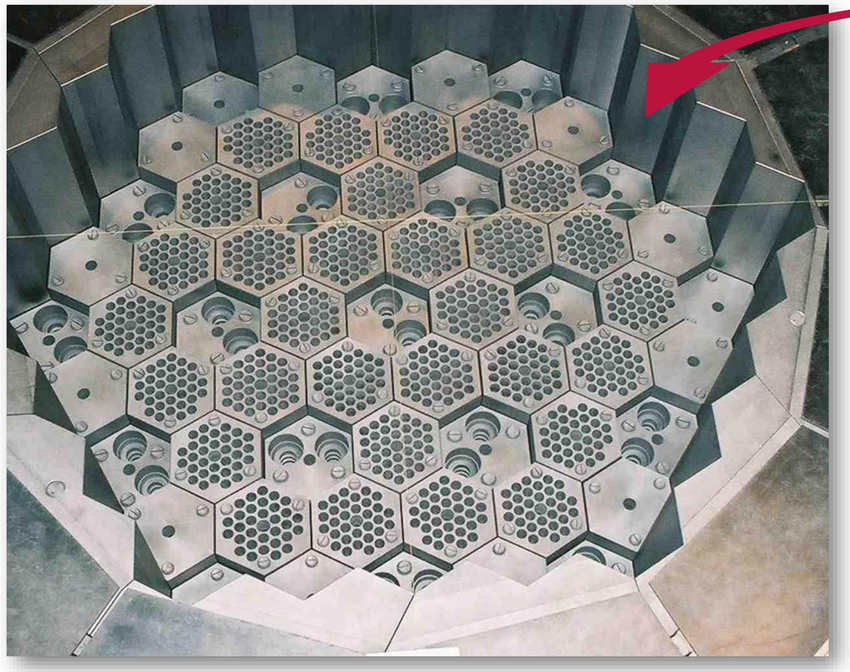
1. The vast majority of fission products in HTGR (prismatic) reactors that appear as a result of fission are retained in the TRISO fuel structure.
2. The retention of the fission products in the TRISO fuel is possible thanks to the carbide coatings surrounding the UO_2.
3. In the event leading to fission products escape from TRISO fuel, most of the material is retained in the primary circuit or in interconnected systems that are isolated from the environment.
4. For most HTR reactors, the only possibility of the release of radioactive substances is their leakage through additional systems such as: the regenerative helium purification system (HPS), or during emergency events - leakage of the primary cooling circuit.

HTGR core



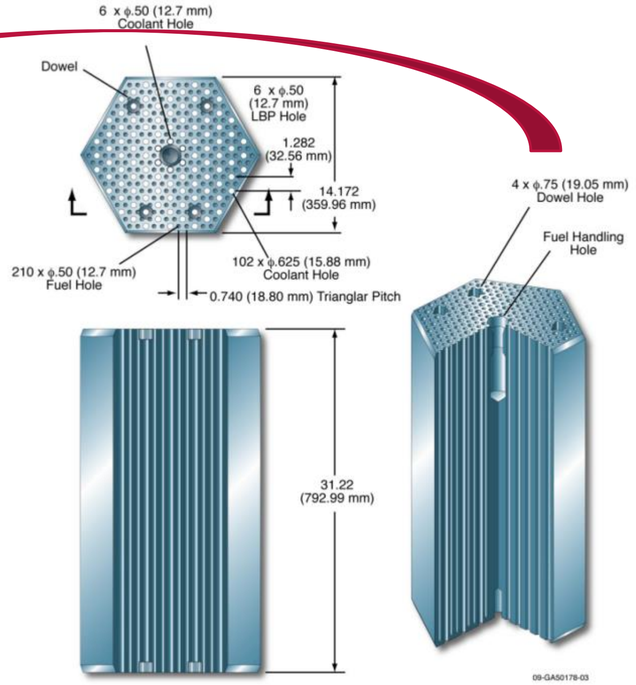
Source: AREVA HTGR High Temperature Gas-cooled Reactor INFORMATION KIT March 2014,High Temperature Gas-Cooled Test Reactor Point Design: Summary Report, J. W. Sterbentz et.al , styczeń 2016, Idaho National Laboratory

Fuel blocks and reactor core



HTTR reactor core

Source: https://japan-forward.com/japan-leads-race-in-development-of-high-temperature-gas-reactors-but-china-is-catching-up/



HTR Fort Saint Vrain fuel block

Source: High Temperature Gas-Cooled Test Reactor Point Design: Summary Report, J. W. Sterbentz et.al , January 2016, Idaho National Laboratory

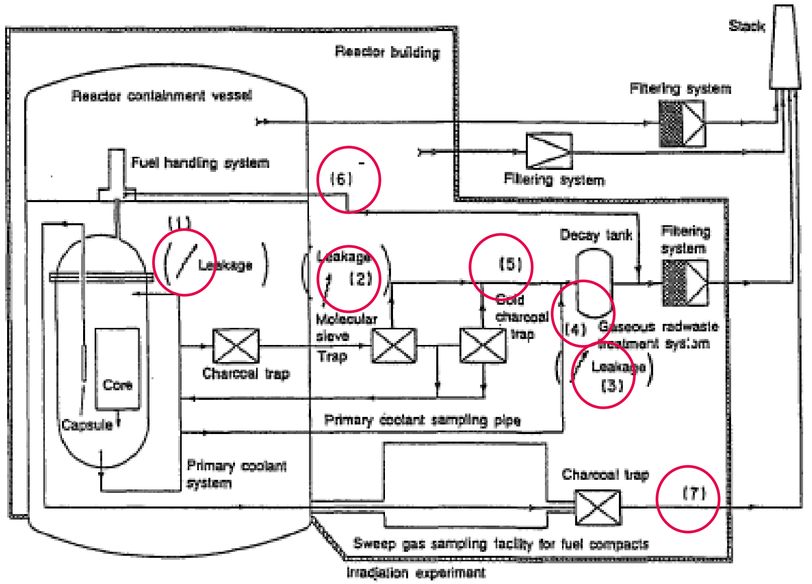
Theoretical and operational releases of HTGR reactors (1/2)

HTTR

- Outlet temperature: 850 °C /950 °C.
- during normal operation, the activity in the cooling circuit is continuously monitored;
- 7 paths of activity release to the environment have been identified
- Radioactive noble gas releases: $3.7 \cdot 10^{13}$ Bq/y, iodine: $3.2 \cdot 10^{9}$ Bq/y, tritium: $1.1 \cdot 10^{13}$ Bq/y

HTR-10

- Outlet temperature: 700°C.
- For the normal operation of the HTR-10 reactor, temperature range for the fuel in the core is: 815.5°C – 864°C, 758.6 to 815.5°C, 644.9 to 758.6°C and below 644.9°C is 0.1, 6.9, 29.4, and 63.7%, respectively.
- The highest release rate from fuel (magnitude of $1 \cdot 10^{8}$ Bq/hMW$_{th}$) – Kr-87, Kr-88, I-132, I-134), long-lived isotopes – $1 \cdot 10^{3}$ Bq/hMW$_{th}$



Environmental release pathways for the HTTR reactor

Source: K. Verfonfdern, J. Sumita, S. Ueat i K. Sawa, „Modeling of fuel performance and fission product release behavior during HTTR normal operation (a comparative study of the FZJ and JAERI modeling approach)," Japan Atomic Energy Research Institute, 2001, W. Von Lensa, G. Brinkmann, J. Lillington, V. Amezcua, O. Hirofumi, M. Futterer i C. Pohl, „Decommissioning and Waste Management of GEMINI system," BriVatech, 2020. L. Yuanzhong i C. Jianzhu, „Fission product release and its environment impact for," Nuclear Engineering and Design, pp. 81-90, March 11 2002.

Theoretical and operational releases of HTGR reactors (2/2)

- HTR-PM – pebble bed, 200 MW_{el}
- Outlet temperature 750°C.
- Total core inventory based on HTR-10 and AVR: 5.12×10^{19} Bq; Kr, Xe – 2.81×10^{18} Bq; Iodine – 2.05×10^{18} Bq; Sr, Cs, Ag 3.05×10^{17} Bq
- Releases to the environment: 5.9×10^{11} Bq noble gases, 5.8×10^{9} Bq iodine, 4.0×10^{5} Bq long-lived isotopes, 8.5×10^{10} Bq tritium, and 1.3×10^{9} Bq C-14.
- HTR-MODULE – pebble bed , 200 MW
- Outlet temperature – 700°C.
- Yearly releases: 7.4×10^{13} Bq/y noble gases, 2.2×10^{7} Bq/y iodine, 3.7×10^{12} Bq/y tritium, 8.9×10^{11} Bq/y C-14.
- AVR – pebble bed, 46 MW_{th}
- Outlet temperature – 850°C.
- Yearly releases: 4×10^{11} Bq/y noble gases, 3.38×10^{4} Bq/y iodine, 2.3×10^{11} Bq/y tritium, 3.8×10^{10} Bq/y C-14.

Source: F. Xie, Y. Liang, J. Zhu, J. Cao i J. Tong, „The Design and Development of the Radiation Monitoring System for the Primary Circuit of HTR-PM," w Proceedings of the HTR 2014, Weihai, China, 2014, G. Strydom, „Reactor Physics Characterization of the HTR Module with UCO Fuel," Idaho National Laboratory, Idaho Falls, Idaho, 2011, K. Kugeler, H. Nabielek i D. Buckthorpe, The High Temperature Gas-coled Reactor, Safety considerations of the (V)HTR-Modul, Luxembourg: Publications Office of the European Union, 2017

Isotopes of interest for the HTGR system

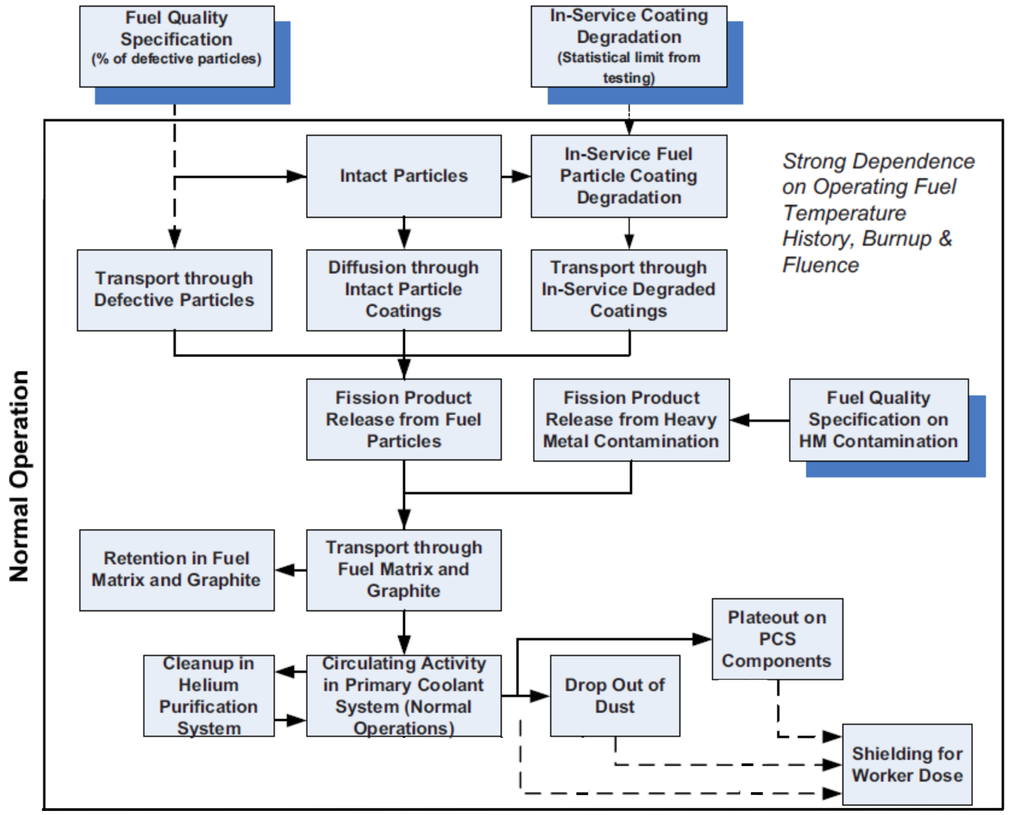
- The TRISO releases rate during normal operation depends on the initial fuel quality and the fuel-related service conditions.
- The initial quality of the fuel and the specification of the failures during operation:
  - determination of the damaged fuel elements population,
  - their intact number and the one that is destroyed during the operation of the reactor.

Radionuclide Class	Key Nuclide	Form in Fuel	Principal In-Core Behavior	Principal Ex-Core Behavior
Tritium	H-3	Element (gas)	Permeates intact SiC; sorbs on core graphite	Permeates through heat exchangers
Noble gases	Xe-133	Element (gas)	Retained by PyC/SiC	Removed by helium purification system
Halogens	I-131	Element (gas)	Retained by PyC/SiC	Deposits on colder metals
Alkali metals	Cs-137	Oxide-element	Retained by SiC; some matrix/graphite retention	Deposits on metals/dust
Tellurium group	Te-132	Complex	Retained by PyC/SiC	Deposits on metals/dust
Alkaline earths	Sr-90	Oxide-carbide	High matrix/graphite retention	Deposits on metals/dust
Noble metals	Ag-110m	Element	Permeates intact SiC	Deposits on metals
Lanthanides	La-140	Oxide	High matrix/graphite retention	Deposits on metals/dust
Actinides	Pu-239	Oxide-carbide	Quantitative matrix/graphite retention	Retained in core

- Radioisotopes that are of interest to HTGR design and safety analysis can be categorized into classes for analysis purposes.
- This division includes radioisotopes within one group, which show similar behavior inside and outside the core (in context of transport modeling).

Source: Idaho National Laboratory, „HTGR Mechanistic Source Terms White Paper," Idaho National Laboratory, NGNP, Idaho Falls, Idaho, 2010.

Radioisotopes in primary circuit



The possible paths of radioactivity releases.

- Isotopes releases and elements in the PCS:
  - Fission products: low fraction of fuel outside carbide coatings ($5x10^{-5}$ g/FE) and fraction of damaged carbide layers during production - $6x10^{-5}$, low fraction of radiation damaged particles - $2x10^{-4}$ (EOC).
    - Noble gases, iodine isotopes (volatile), highly volatile solid isotopes - Sr90, Cs134, Cs137 i Ag110m .
  - Tritium – can penetrate metallic pipes to the secondary side.
  - C-14 – neutron capture in N_2 impurities of the core structures.
  - Corrosion products – small, thanks to the helium characteristics.

Source: Idaho National Laboratory, „HTGR Mechanistic Source Terms White Paper," Idaho National Laboratory, NGNP, Idaho Falls, Idaho, 2010, D. V. A. J. Gerd Brinkmann, „Final GEMINI + Safety Options Report," European Union's Horizon 2020 Research and Innovation Program, 2020.

Releases during normal operation – primary circuit

- Liquid waste – contained before utilization (around 5 GBq/y for 180MWth NPP)
- Additional systems in PCS
  - Helium purification system – ~5%/h of flow
  - Regeneration system – cyclical operation
  - Gas removal system from helium purification system
  - System for gas discharge from the primary cooling circuit
  - Helium discharge system for support systems
- Stabilization of the circulating activity in the primary circuit (except long-lived isotopes).
  - fuel releases,
  - the half-life of radioisotopes,
  - absorption of radioisotopes in fuel, graphite matrix and graphite dust,
  - deposition of radioisotopes on the surfaces of the components of the primary cooling circuit,
  - fall of radioisotopes in places with low flow velocity,
  - and the efficiency and operation mode of the helium coolant purification system (HPS).

Steady state coolant activity in Bq (magnitude)

(relative to 8800m^3 of Helium (20°C, 1 bar)

Total iodine	10^9	
Total noble gases	10^{12}	Kr, Xe
Total short-lived solids	10^{10}	Rb88, Cs138
Total long-lived solids	10^6	Sr90, Cs134, Cs137, AG110m
Tritium	10^{12}	
C14	10^{10}	

Surface activity after 54 EFPY in Bq (magnitude)

Total iodine	10^{12}	Kr, Xe
Total short-lived solids	10^{12}	Rb88, Cs138
Total long-lived solids	10^{13}	Sr90, Cs134, Cs137, AG110m

Radioactive material in the primary circuit and on the primary circuit surfaces for the GEMINI + reactor

Source: D. V. A. J. Gerd Brinkmann, „Final GEMINI + Safety Options Report," European Union's Horizon 2020 research and innovation programme, 2020.

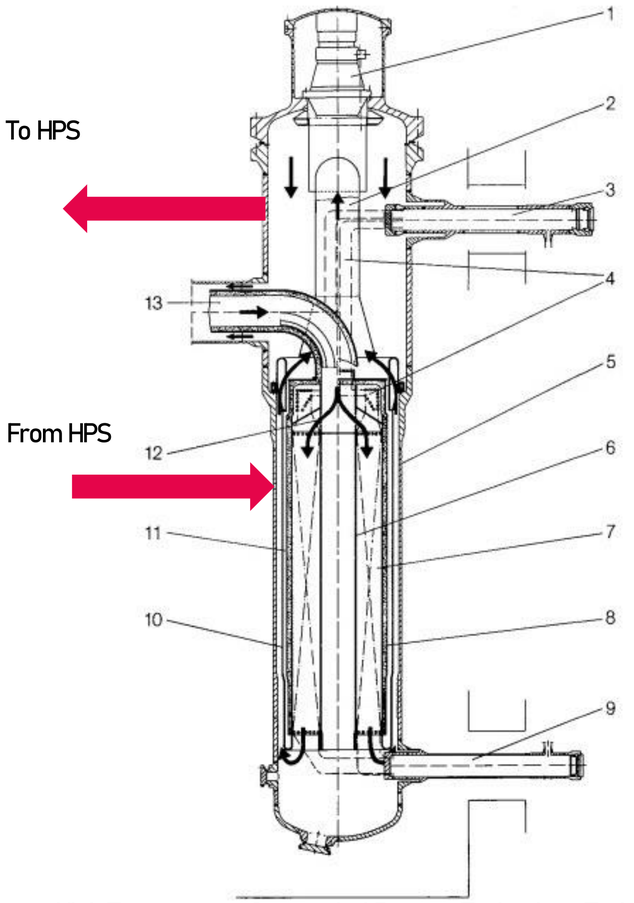
Helium Purification System

Functions:

- Removes chemical contaminants and radionuclides from helium coolant.
- Increases pressure, lowers pressure and controls helium primary coolant in conjunction with the helium transfer and storage system.
- Keeps the main coolant system pressure below atmospheric one during fuel reloading / maintenance.
- It cleans the helium pumped into the storage tanks.
- Removes H_2O from the primary circuit after water ingress.

Characteristics:

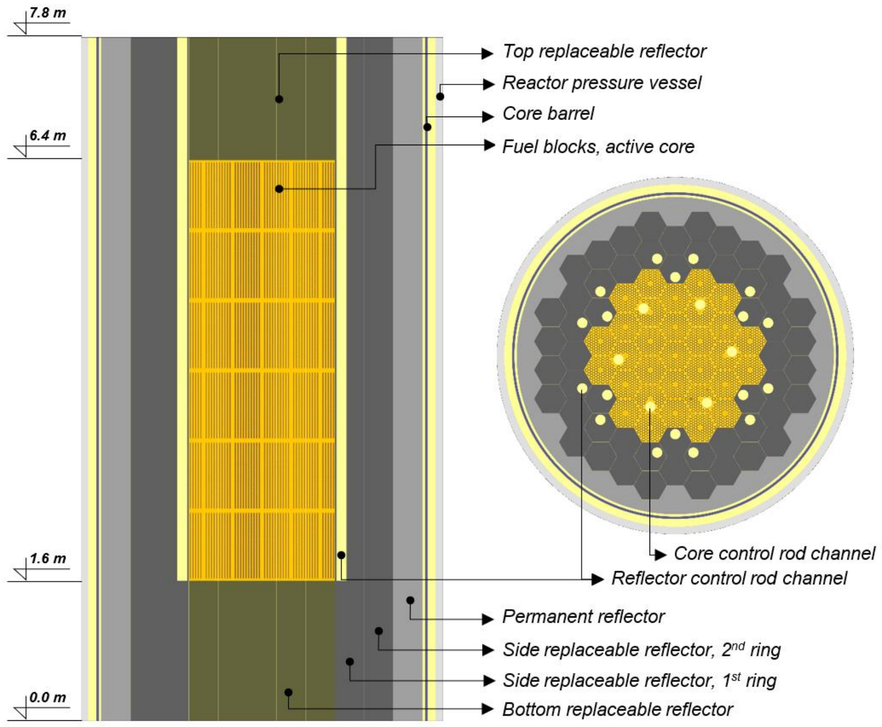
- Removes: H_2O, CO, CO_2, H_2, N_2, O_2, H_2S, CH_4, Tritium, Xe, Kr, Ar.
- Cyclical operation: cleaning and regeneration cycle (after consuming the substrates).
- Redundant system (two lines).
- It works with an efficiency - about 1% - 5% of the primary circuit volume flow.
- Uses copper oxide deposits (HT, H2 and CO oxidation), adsorption sieves - the first "hot" stage.
- Uses adsorptive activated carbon (Kr, Xe, Ar, N2, CH4) - second stage "cold„.



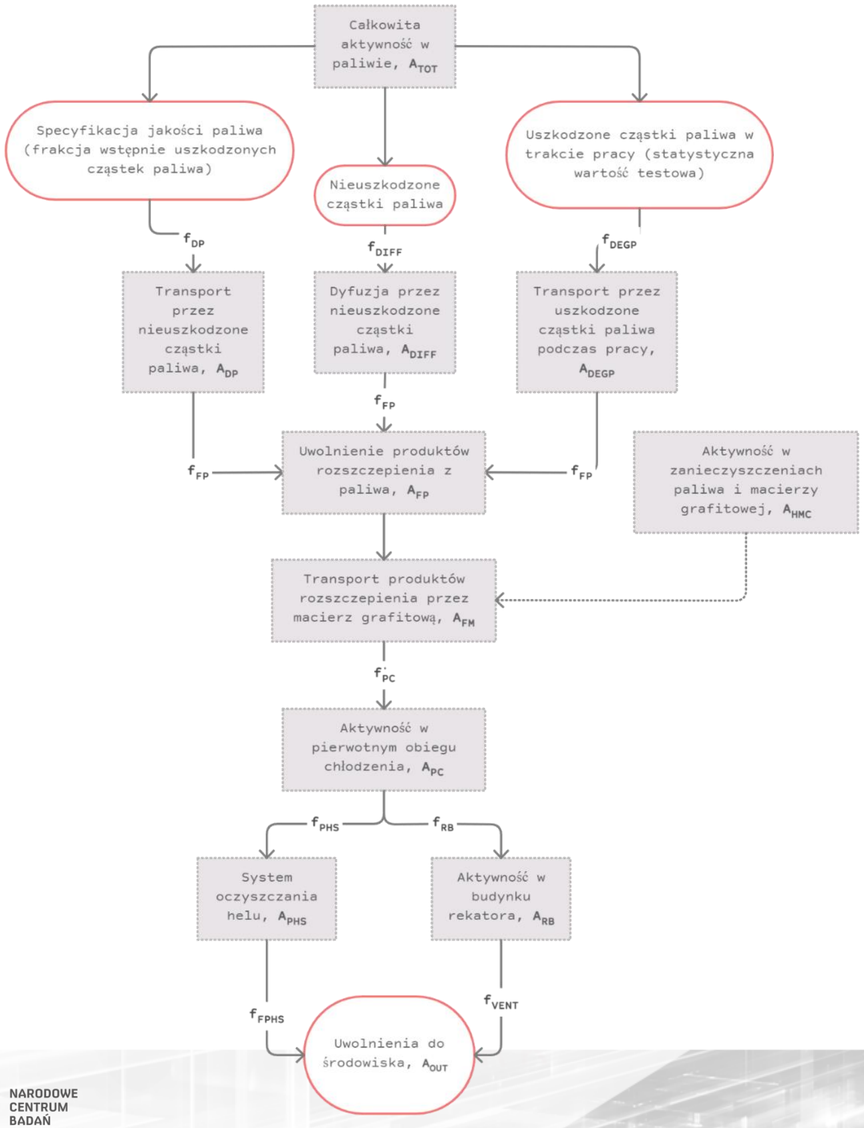
Source: High Temperature Gas-cooled Reactor Technology Training Curriculum, Presented by Idaho National Laboratory, July 16-17, 2019, D2.7 Final GEMINI + Safety Options Report, 2020, The High Temperature Gas-cooled Reactor Safety considerations of the (V)HTR-Modul, Kugeler, K., Nabielek, H., Buckthorpe, D., The helium purification system of the HTR-10, M.S.Yao, R.P.Wang, Z.Y.Liu, X.D.He, J.Li

TeResa core

- Technology demonstrator proposal - NCBJ implementation.
- Development of a preconceptual design of a 40 MW_{th} reactor.
  - Neutron analysis of the cycle - enrichment, fuel packing in compacts, power distribution, burnup.
  - Thermal-hydraulic analyzes - compliance with the requirements set out by regulations and best international practices.
- Core height – reduced by 1/2 relative to GEMINI + design.
- Maintaining the diameter of the tank and the heat removal system from the RPV.
- Demonstration of the correct operation of the system.
- Preparation of a partial safety report for the "research" reactor - preparation of a safety assessment methodology for HTGR installations in Poland.



TeResa reactor data



Release paths:

a. Air activation in the reactor cavities.

b. Leaks from the PCS and attached systems.

c. Leaks from the radioactive substances storage tank.

d. Leaks related to the maintenance and fuel reloading operations.

Due to unwanted character of points c and d only a and b are taken into consideration.

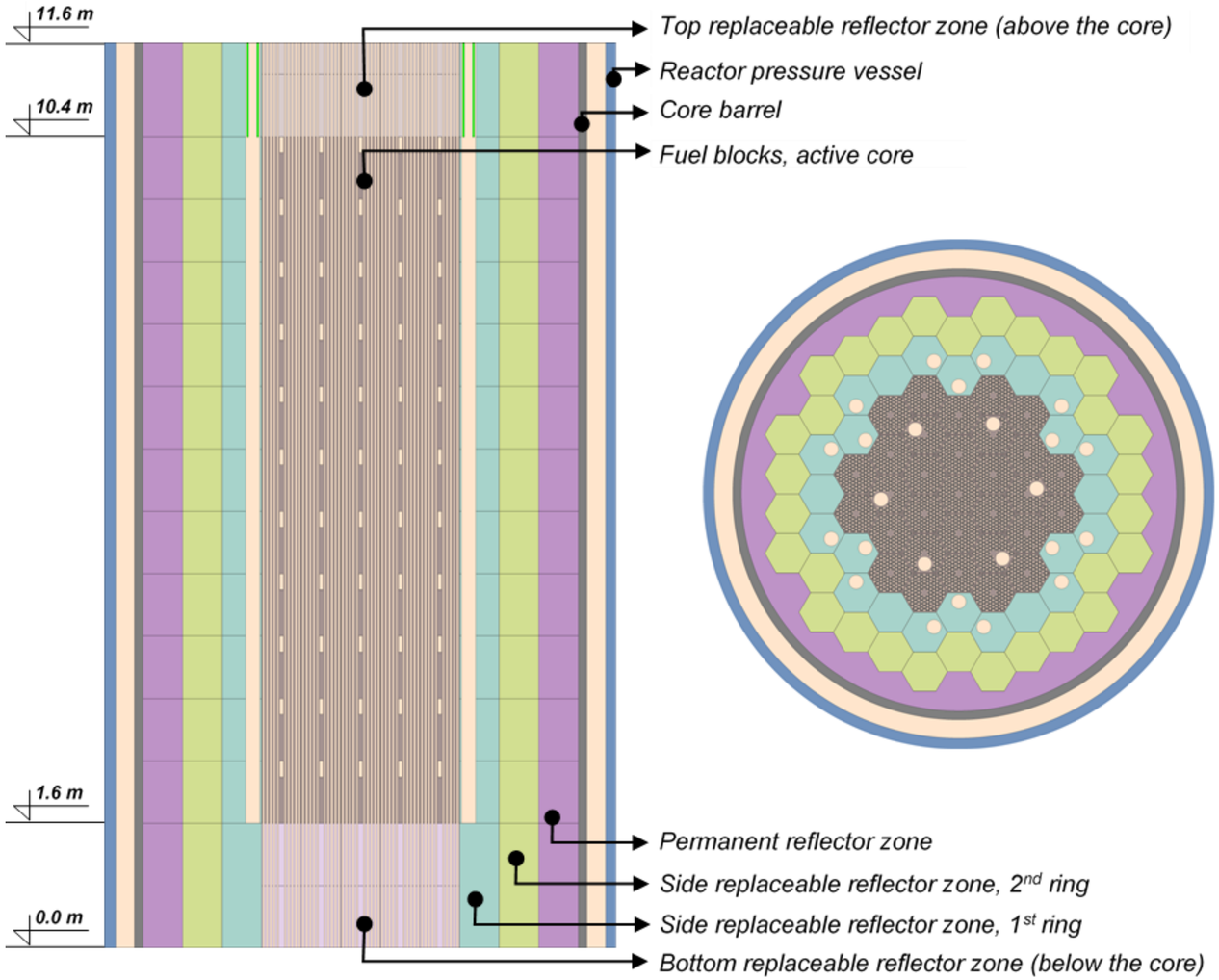
Ad. a. Air activation by neutron capture occurs only in the volumes near the reactor pressure vessel (Ar-41).
Sub-atmospheric RB pressure - at most 2000 m^3 of air per day is extracted from the volumes surrounding the reactor vessel and transferred to the ventilation stack.

Ad. b. PCS leakage - occurs in the form of noble gases (also aerosols and iodine isotopes).
A total primary coolant leakage of 1/1000 is assumed from the primary coolant transfer systems, with an average air exchange rate of 1/h for rooms prone to these leaks.

Source: D2.7 Final GEMINI + Safety Options Report, 2020

Data needed for the releases estimations

Input data



Radionuklid	T_1/2		Xe135m	15.29	M
C14	5700	Year	I131	8.0252	D
H3	12.32	Y	I132	2.295	H
Kr83m	1.83	Hour	I133	20.8	H
Kr85	10.756	Y	I134	52.5	M
Kr85m	4.48	H	I135	6.58	H
Kr87	76.3	Minute	Rb88	17.773	M
Kr88	2.84	H	Sr89	50.53	D
Xe131m	11.84	Day	Sr90	28.79	Y
Xe133	5.243	D	Cs134	2.0652	Y
Xe133m	2.19	D	Cs137	30.08	Y
Xe135	9.14	H	Ag110m	249.76	D

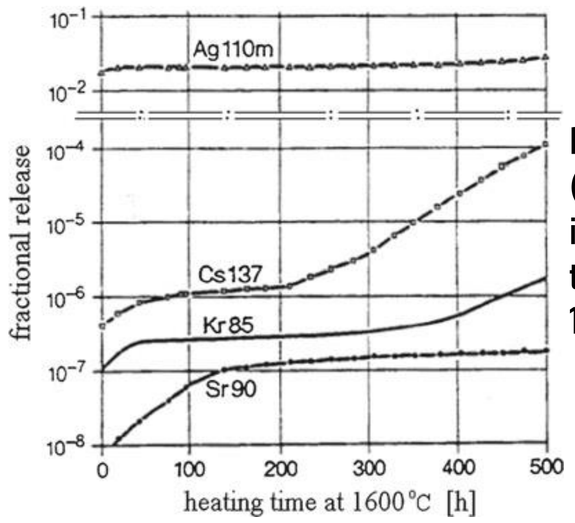
- Calculations performed with Serpent neutronic (MC) code.
- Burnup calculations made for 550 days.
  - Fuel, graphite blocks, burnable poisons
  - 2 cycles for bottom and top, 2 rings of radial reflector.
  - Never for permanent reflector.

Source: J. Kuijper i D. Muszyński, „Core design calculations of GEMINI + system," GEMINI+, 2021

Simplified parametric model

Methodology

- The example methodology (used also for RDE reactor (Indonesia)), that results in simple nuclear reactor release calculations:
  - ❑ based on the analysis of possible release paths (slide 13) and
  - ❑ release rates from the components. The fuel release rates and other factors related to the transport of radioactive isotopes in the graphite matrix and coolant selected from the literature.



Releases (fractions) for isotopes in temperature of 1600 °C

Prismatic Barrier AFs During Normal Operations

Fission Product Class	Heavy Metal Contamination		Fuel Particle Kernel		Diffusive Release Through Fuel Particle Coatings		Graphite (Compact Matrix and Fuel Element)		Helium Pressure Boundary	
Confidence Limit	AF_{HMC} 50%	AF_{HMC} 95%	AF_K 50%	AF_K 95%	AF_{Diff} 50%	AF_{Diff} 95%	AF_G 50%	AF_G 95%	AF_{HPB} 50%	AF_{HPB} 95%
700°C Reactor Outlet Temperature										
Noble gases	10	3	50	17	1E8[a]	1E7	1	1	1	1
I, Br, Se, Te	10	3	50	17	1E8	1E7	1	1	1E6	1E5
Cs, Rb	1	1	3	1	1E8	1E6	5	2	1E6	1E5
Sr, Ba, Eu	1	1	50	20	1E3	200	1E3	300	1E6	1E5
Ag, Pd	1	1	2	1	500[b]	100[b]	2	1	1E6	1E5
Sb	1	1	2	1	1E8	1E6	20	2	1E6	1E5
Mo, Ru, Rh, Tc	1	1	500	30	1E8	1E7	1E3	300	1E6	1E5
La, Ce	1	1	500	30	1E8	1E7	1E3	300	1E6	1E5
Pu, actinides	1	1	1E3	100	1E8	1E7	1E4	1E3	1E6	1E5
900°C Reactor Outlet Temperature										
Noble gases	5	1.5	25	8.33	5E7	5E6	1	1	1	1
I, Br, Se, Te	5	1.5	25	8.33	5E7	5E6	1	1	1E6	1E5
Cs, Rb	1	1	1.2	1	1E7	1E5	2	1	1E6	1E5
Sr, Ba, Eu	1	1	3	1	500	100	100	30	1E6	1E5
Ag, Pd	1	1	1	1	200[b]	40[b]	1	1	1E6	1E5
Sb	1	1	1	1	5E7	5E5	5	1	1E6	1E5
Mo, Ru, Rh, Tc	1	1	250	15	1E7	1E6	200	60	1E6	1E5
La, Ce	1	1	250	15	1E7	1E6	200	60	1E6	1E5
Pu, actinides	1	1	500	50	1E7	1E6	5E3	500	1E6	1E5

[a]Read as 1×10^8.
[b]Values presented here for ^{110m}Ag. For ^{111}Ag, the values for the diffusive release through the coating are increased by a factor of 5 to account for the effect of the half-life on the release.

Source: K. Verfonfdern, J. Sumita, S. Ueat and K. Sawa, „Modeling of fuel performance and fission product release behavior during HTTR normal operation (a comparative study of the FZJ and JAERI modeling approach),” Japan Atomic Energy Research Institute, 2001, J. Cao, L. Zhang, F. Xie, B. Xia and S. Tsz Tang Lam, „Source Term Study on Tritium in HTR-PM: Theoretical Calculations and Experimental Design,” Science and Technology of Nuclear Installations, Lipiec 2017, : D. A. Petti, R. R. Hobbins, P. Lowry & H. Gougar (2013) Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors, Nuclear Technology, 184:2, 181-197, DOI: 10.13182/NT184-181, L. Yuanzhong and C. Jianzhu, „Fission product release and its environment impact for,” Nuclear Engineering and Design, pp. 81-90, 11 marzec 2002,
Z.Liu , J. Cao, „Fission Product Release and Its Environment Impact for Normal Reactor Operations and for Relevant Accidents,” Nuclear Engineering and Design, 2002,
Idaho National Laboratory, „HTGR Mechanistic Source Terms White Paper,” Idaho National Laboratory, NGNP, Idaho Falls, Idaho, 2010,
Udiyani i S. Kuntjoro, „Estimation Of Routine Discharge Of Radionuclides On Power Reactor Experimental RDE,” Urania, pp. 1-68, Luty 2017.

Simplified parametric model

Methodology

Releases from fuel:

$$A_{i,FP} = \left(A_{i,DP} + A_{i,DIFF} + A_{i,DEGP}\right) \quad (1)$$

$$A_{i,DP} = f_{FP} \cdot A_{i,TOT} \quad (2)$$

$$A_{i,DIFF} = f_{DIFF} \cdot A_{i,TOT} \quad (3)$$

$$A_{i,DEGP} = f_{DEGP} \cdot A_{i,TOT} \quad (4)$$

i – isotope;

$A_{i,TOT}$ – Total fuel activity in the active core area [Bq];

$A_{i,DP}$ – Activity released from pre-damaged fuel particles [Bq];

$A_{i,DIFF}$ – Activity released from fuel via a fission product diffusion mechanism [Bq];

$A_{i,DEGP}$ - Activity released from damaged fuel particles during irradiation [Bq];

f_{FP} – the release rate of the fission products from the fuel particles;

f_{DP} - the ratio of pre-damaged fuel particles;

f_{DIFF} - the rate of diffusion-mediated fission products from undamaged fuel particles;

f_{DEGP} - damaged particle ratio of fuel during irradiation.

Simplified parametric model

Methodology

Activity in the graphite matrix:

$$A_{i,FM} = A_{i,FP} + A_{i,HMC} + A_{i,FSC} \quad (5)$$

$A_{i,FM}$ – activity released from the graphite matrix [Bq];

$A_{i,FP}$ – activity released from fuel [Bq];

$A_{i,HMC}$ – activity released from the heavy elements contamination of the graphite matrix [Bq];

$A_{i,FSC}$ – aktywność uwolniona z paliwa z zanieczyszczeń [Bq].

Transport of fission products to the environment (release):

$$A_{i,OUT} = A_{i,PHS} + A_{i,RB} \quad (6)$$

$$A_{i,PC} = f_{i,PC} \cdot A_{i,FM} \quad (7)$$

$$A_{i,PHS} = A_{i,PC} \cdot f_{i,PHS} \cdot f_{i,PHT} \quad (8)$$

$$A_{i,RB} = A_{i,PC} \cdot f_{i,RB} \cdot f_{i,VS} \quad (9)$$

$A_{i,PC}$ – activity released into the primary cooling circuit [Bq];

$A_{i,RB}$ – activity released into the reactor building [Bq];

$A_{i,PHS}$ – activity released into the helium purification system [Bq];

$f_{i,PC}$ – the fraction of fission products circulating in the primary cooling circuit;

$f_{i,RB}$ – air removal rate to the reactor building;

$f_{i,PHS}$ – the ratio of the gas flowing into the helium purification system;

$f_{i,VS}$ – air removal rate from the reactor building to the environment;

$f_{i,PHT}$ – the ratio of the released gas from the helium purification system to the environment.

Estimation of the Tritium, C-14 and Ar-41 releases

- The estimation of the isotopes of tritium, carbon (C-14) and argon (Ar-41):
  - limited input data,
  - carried out on the basis of literature data and the data of operating reactors (Peach Bottom, HTR-10 and AVR).
  - depending on the characteristics of the isotope and its origin, scaling to the power of TeResa reactor was performed using the factors listed below.

Peach Bottom (66.5 MWth) (measured in coolant): 5.439E+11 Bq/y

$f_{scaling}$ ($P/V_{coolant}$)= 0.6015

TERESA H3 coolant: 3.2716E+11 Bq/y

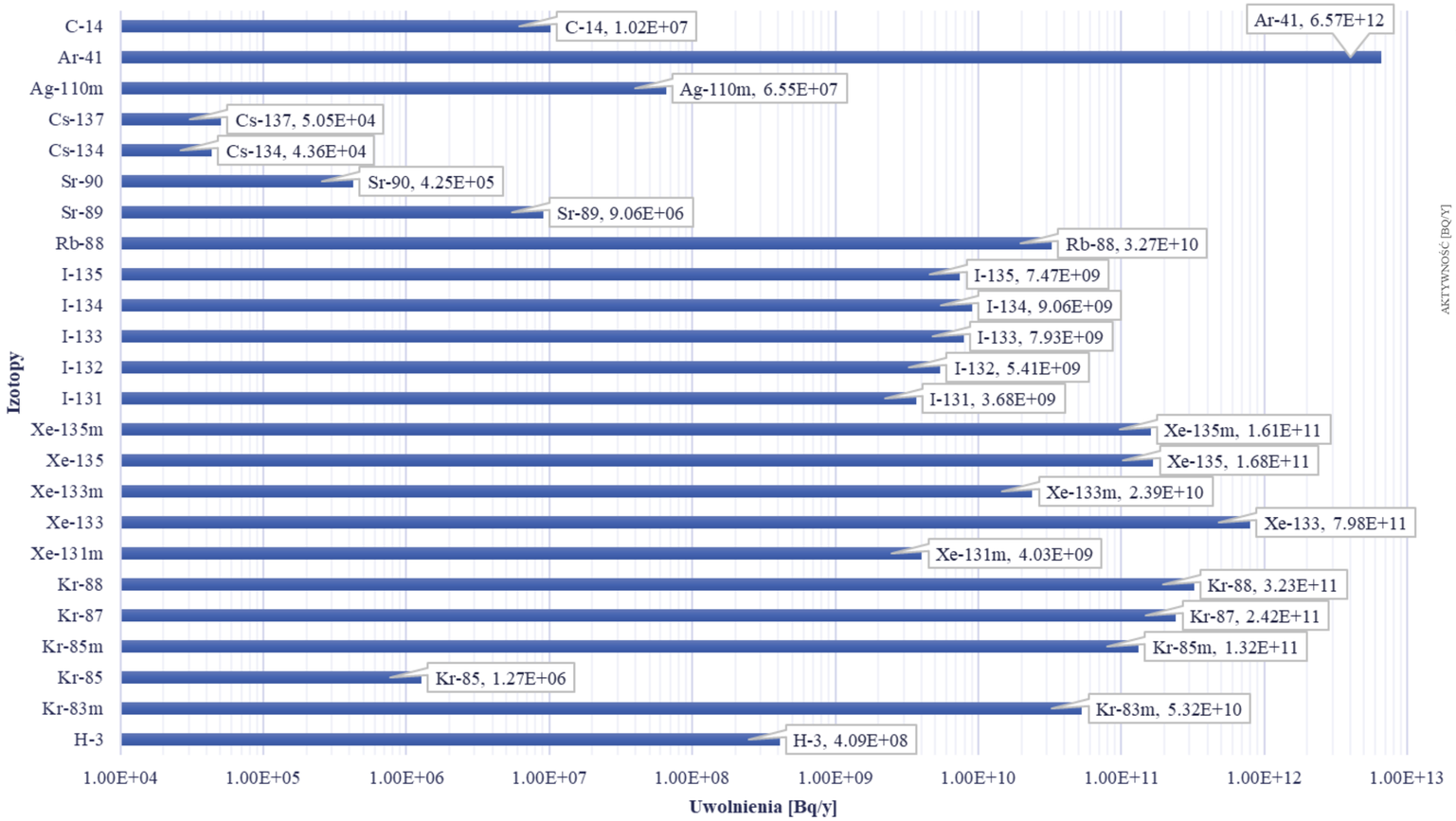
	Designed [Bq/y]	Calc. [Bq/y]
Releases of HTR-10 Ar-41	4.10E+12	4.52E+12
V_{TERESA}/V_{HTR-10} (H_T/H_{HTR-10})	1.455	
Releases TERESA Ar-41	6.57E+12 [Bq/y]	

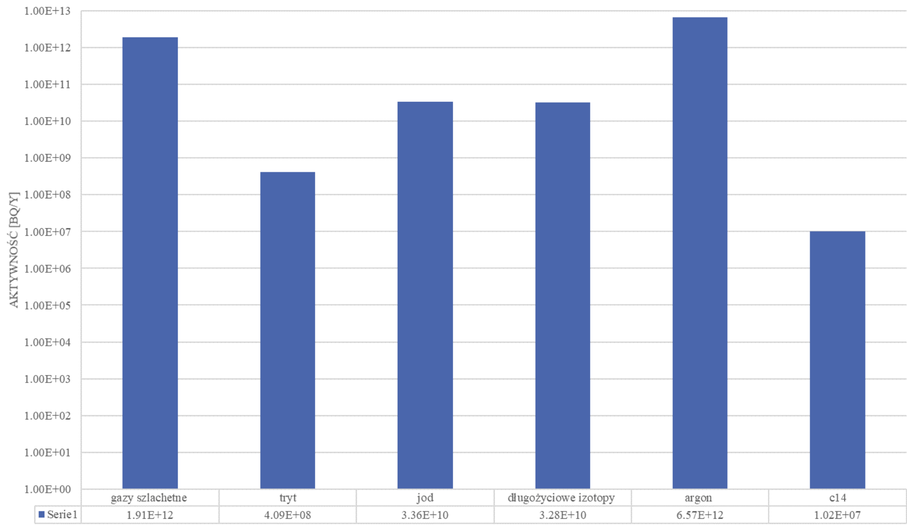
C-14 AVR coolant inventory 1.90E+07 Bq/m^3

The experience gathered in the operation of the AVR, taking into account the relationship between the power and the production of C-14, is the value of 100,000 Bq / hMW.

No filtration results

Results



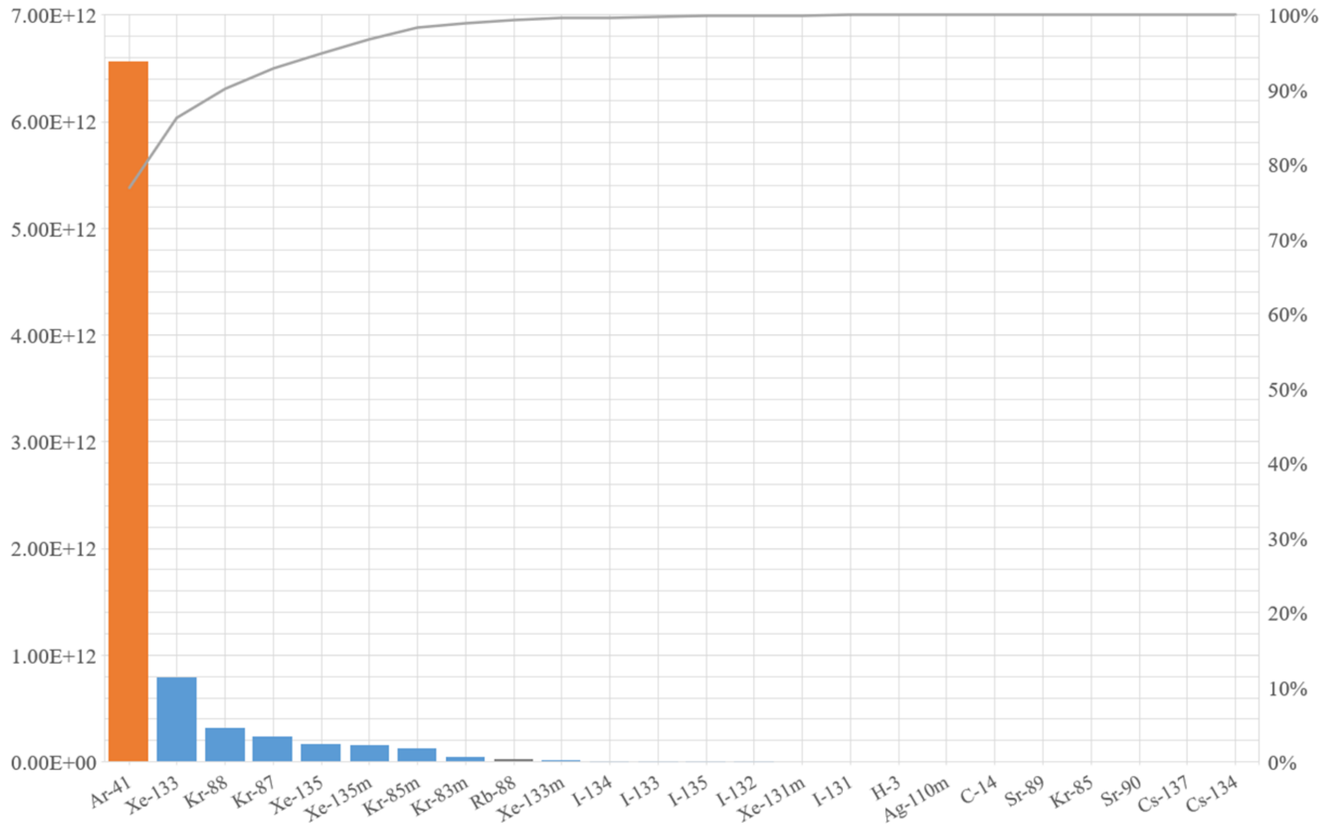


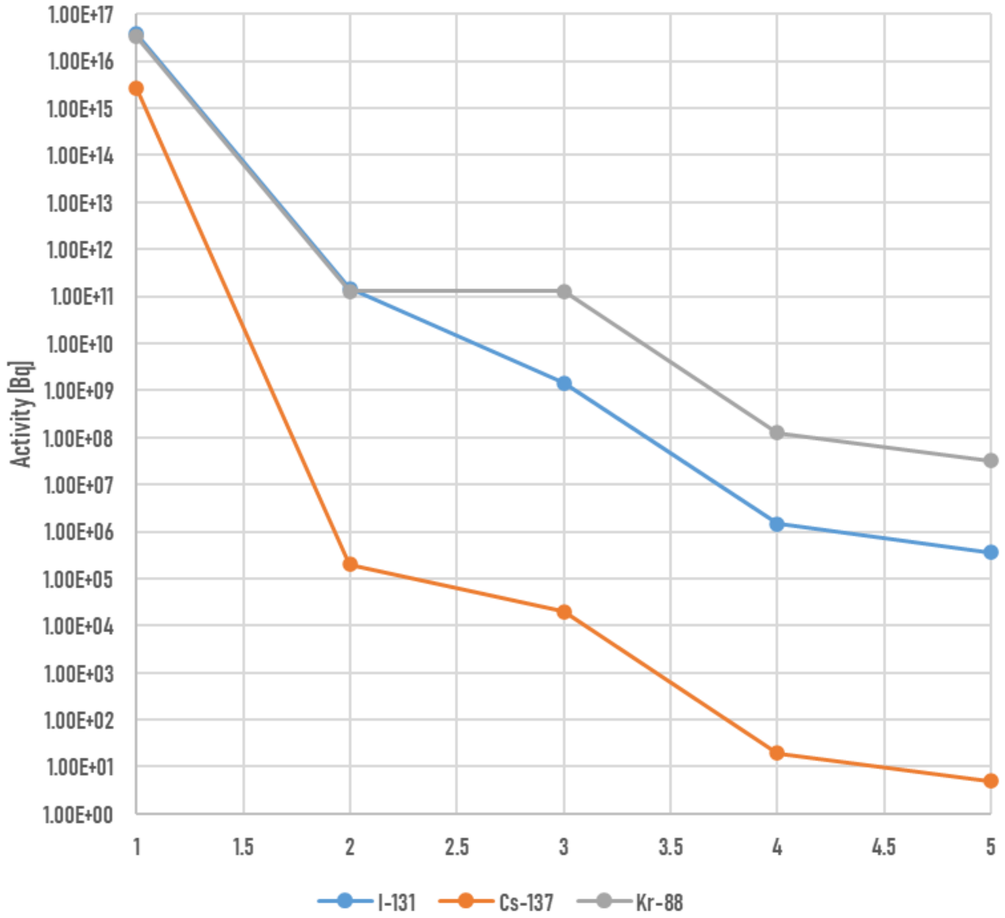
Noble gases	1.91E+12
Tritium (core born)	4.09E+08
Iodine	3.36E+10
Long-lived isotopes	3.28E+10
Argon	6.57E+12
C-14	1.02E+07

Yearly limit NG and Ar - 0.867%

No filtration results

Results

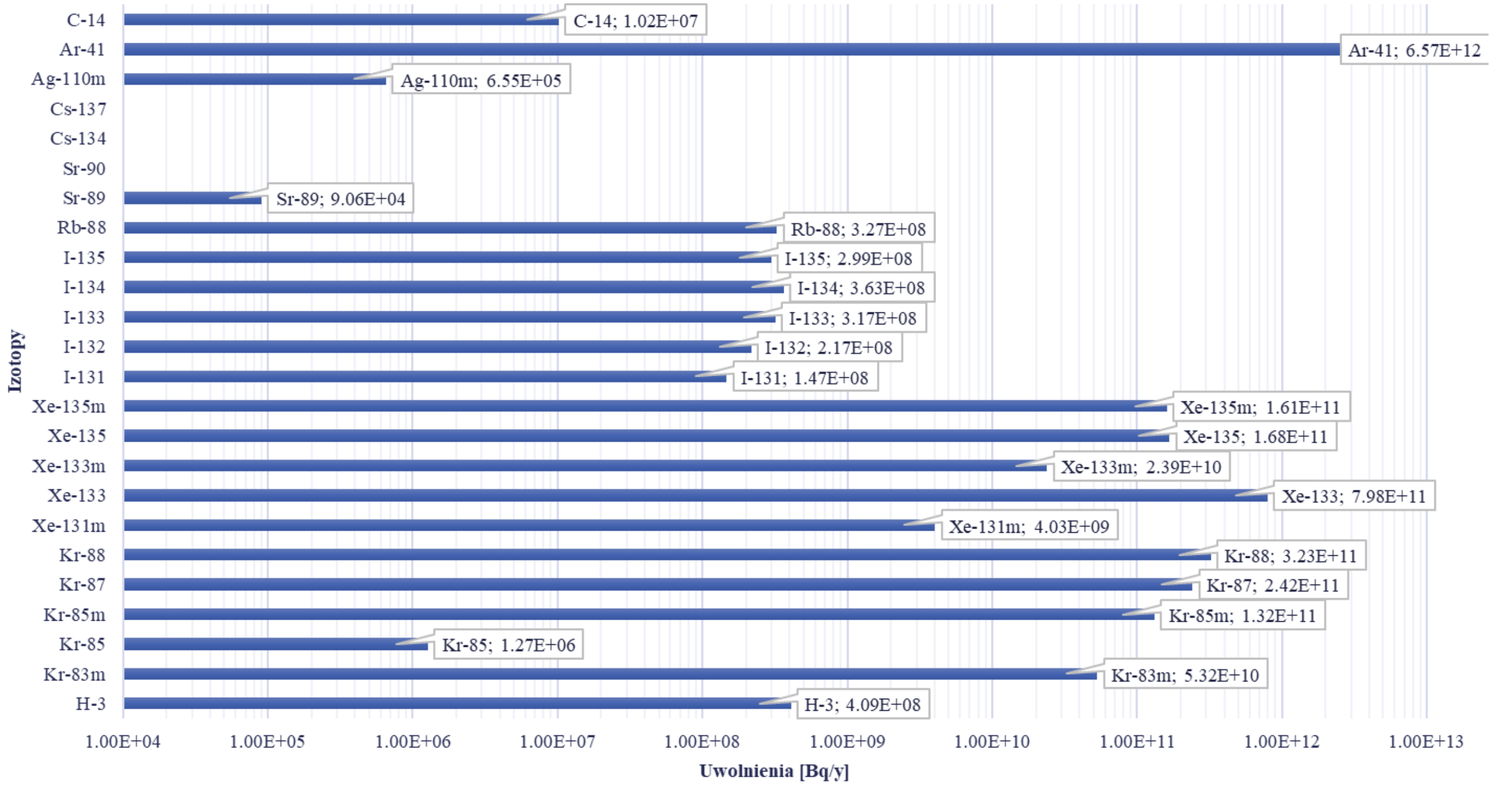


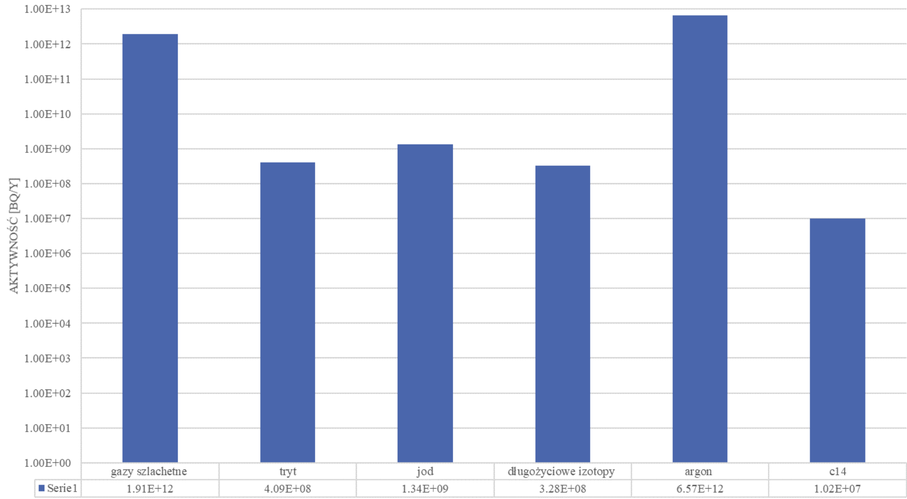


1. Activity of inventory (Bq) 2. Transport through fuel matrix and graphite 3. Circulating activity in primary coolant 4. Release to the reactor building 5.Release to the helium purification system

Added filtration results

Results





Noble gases	1.91E+12
Tritium (core born)	1.77E+10
Iodine	1.34E+09
Long-lived isotopes	3.28E+08
Argon	6.57E+12
C-14	1.02E+07

Filtration fraction – no result for noble gases, 96% for iodine and 99% for long-lived isotopes.

Yearly limit Iodine - 27%

Alternative methodology

Alternative

- Based on the MHTGR studies by Idaho National Laboratory.
- Different release fractions from fuel and plate-out phenomena taken into account.

$$R^i_{\mathrm{HMC}} = \frac{Inv^i * \mathrm{HMC}}{\mathrm{AF}^i_{\mathrm{HMC}} * \mathrm{AF}^i_G}\ , \qquad (1)$$

R^i_{HMC} = release of fission product i from heavy metal contamination (curies)

Inv^i = inventory of fission product i

HMC = level of heavy metal contamination (see Table III)

$\mathrm{AF}^i_{\mathrm{HMC}}$ = attenuation factor of fission product i for heavy metal contamination (see Table V)

AF^i_G = attenuation factor of fission product i in graphite (see Table V).

$$R^i_{DSiC+ISF} = \frac{Inv^i * (DSiC + ISF)}{\mathrm{AF}^i_K * \mathrm{AF}^i_G}\ , \qquad (2)$$

where

$R^i_{DSiC+ISF}$ = release of fission product i from SiC defects and in-service failures (curies)

Inv^i = inventory of fission product i

$DSiC$ = level of SiC defects (see Table III)

ISF = level of in-service failures (see Table IV)

AF^i_K = attenuation factor of fission product i in kernel (see Table V)

AF^i_G = attenuation factor of fission product i in graphite (see Table V).

$$R^i_{Diff} = \frac{Inv^i}{\mathrm{AF}^i_{Diff} * \mathrm{AF}^i_G}\ , \qquad (3)$$

where

R^i_{Diff} = diffusive release of fission product i (curies)

Inv^i = inventory of fission product i

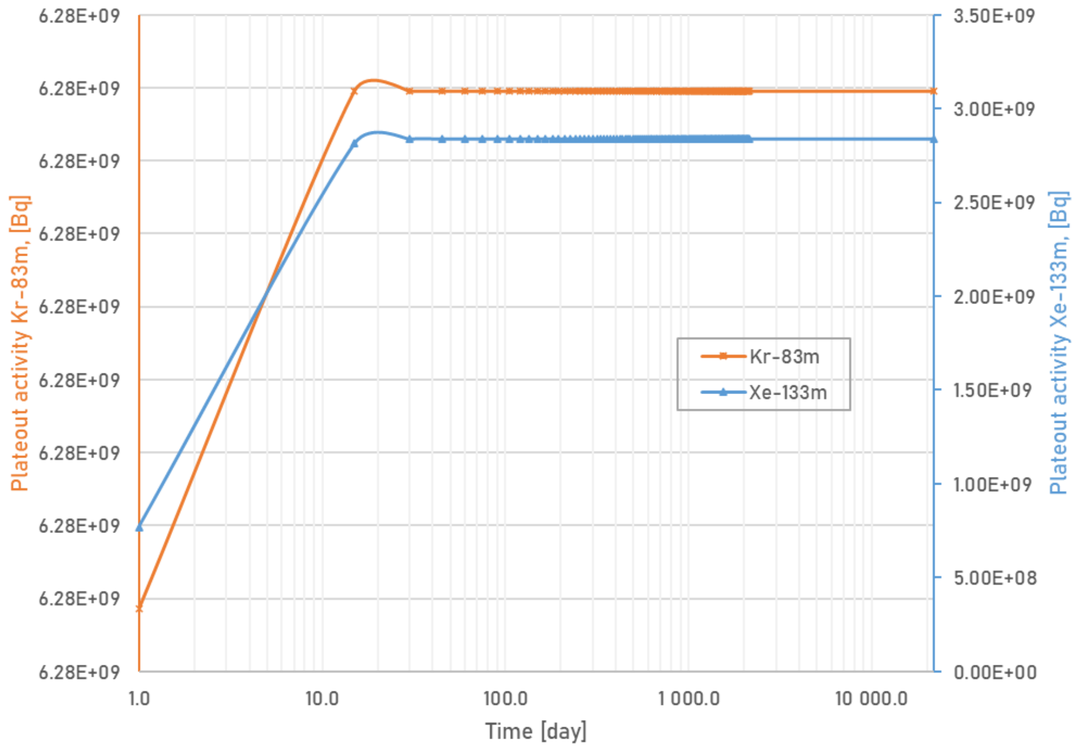
AF^i_{Diff} = attenuation factor of fission product i in intact particle coatings (see Table V)

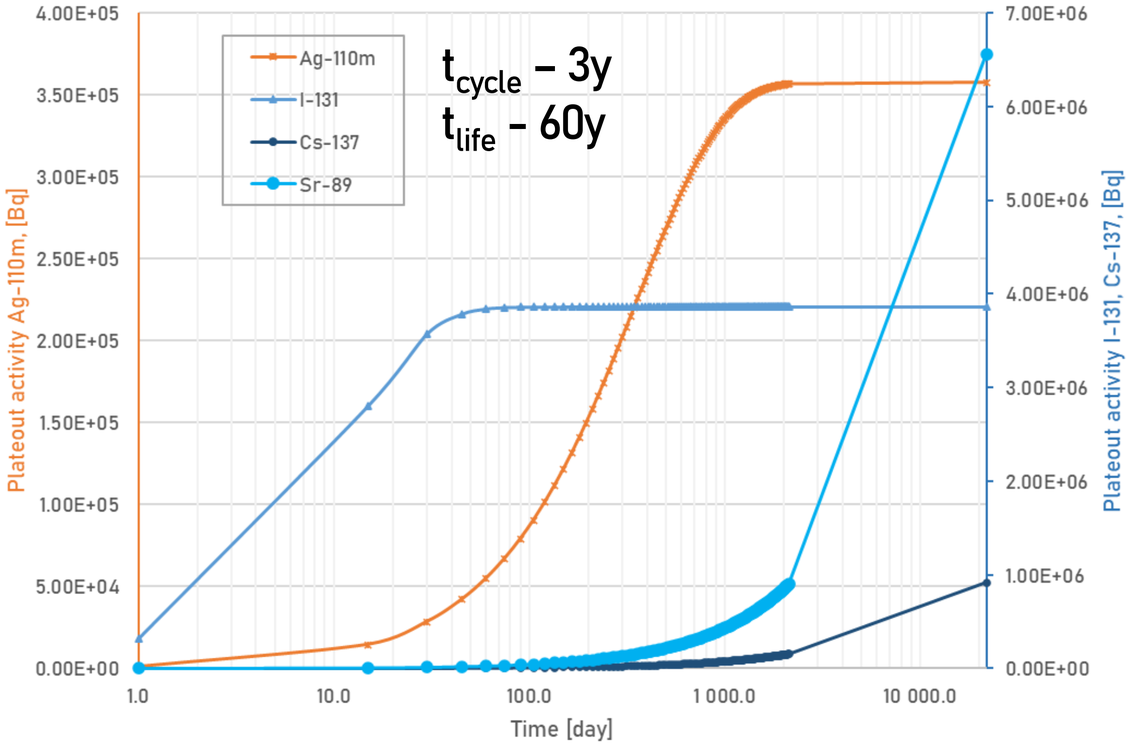
AF^{Diff}_G = attenuation factor of fission product i in graphite (see Table V).

Source: D. A. Petti, R. R. Hobbins, P. Lowry & H. Gougar (2013) Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors, Nuclear Technology, 184:2, 181-197, DOI: 10.13182/NT184-181

Plateout effects

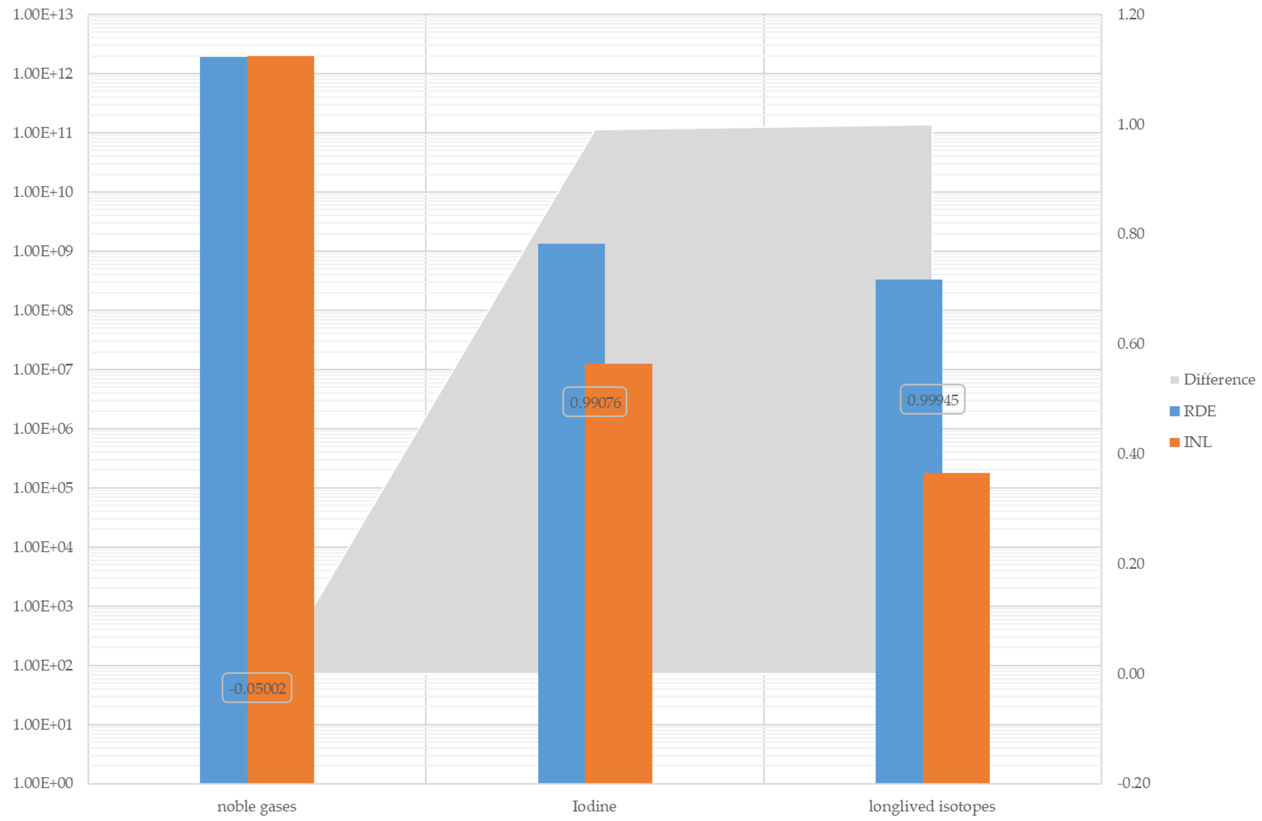
Alternative

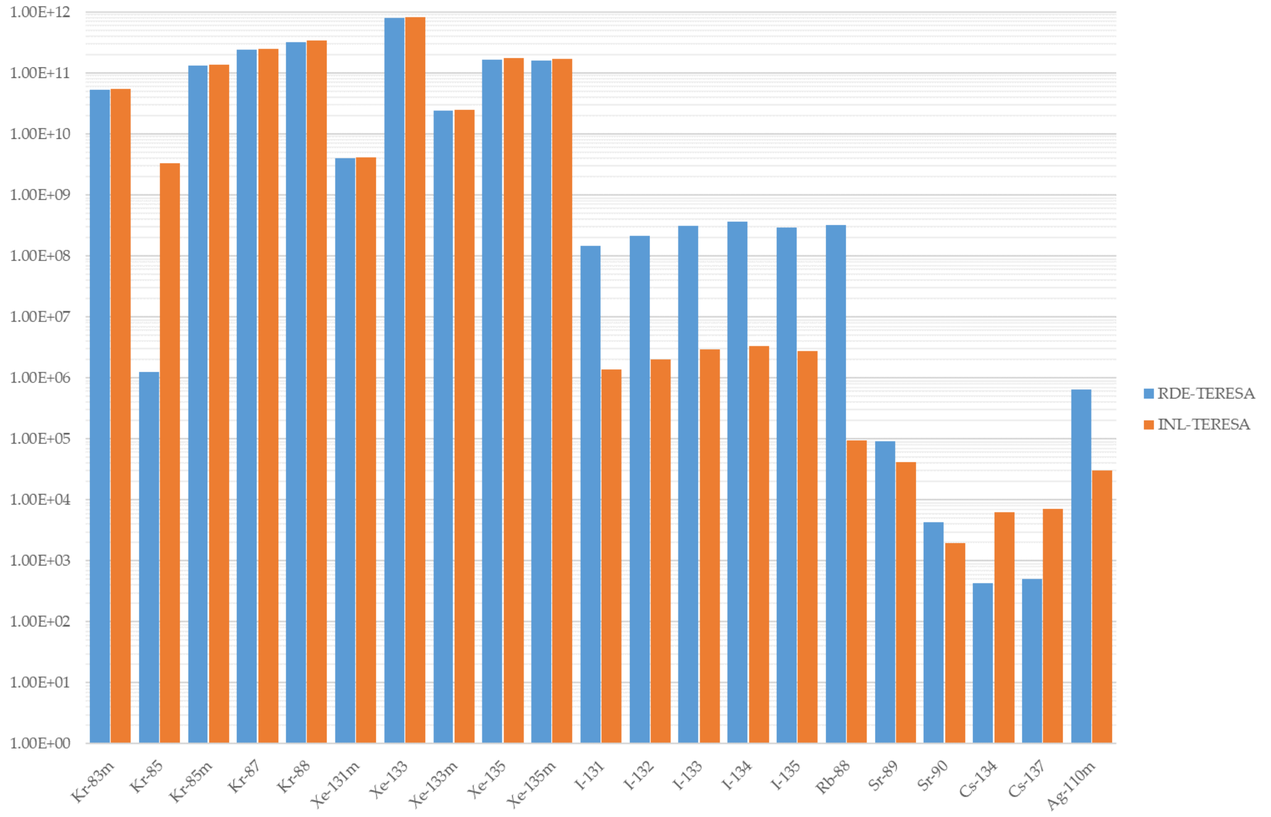




$$PltInv^{i} = R^{i}_{PC} \cdot \frac{1 - e^{-\lambda tlife}}{1 - e^{-\lambda tcycle}}$$

Comparison methodology RDE and INL





Summary

- The analysis of radioactive isotope releases was carried out for the TeResa reactor on the basis of data and experience from the GEMINI + project.
- The methodology that was adopted from literature is based on a simplified modeling of releases with assumptions of fission product retention factors for individual components of the reactor system.
- Assumptions of the fuel release factors for selected radioactive isotopes were made on the basis of the literature and experimental data for TRISO fuel, and the core activity values were scaled from the GEMINI + core in relation to the reactor power ratio.
- Based on the operational and computational experience of other HTR reactors (AVR, Peach Bottom and HTR-10), the values of the releases of tritium (H3), carbon (C14) and argon (Ar41) were estimated – $4.09 \cdot 10^{8}$ Bq / y, $6.57 \cdot 10^{12}$ Bq / y and $1.0 \cdot 10^{7}$ Bq / y.
- The results presented are given for the ventilation system for the TeResa reactor, without a filtration system and taking into account systems adapted from the MARIA reactor. The filtering system has a significant impact on the values of isotope releases from the iodine group and long-lived solid isotopes removed from the reactor building and helium purification system – filtering coefficients were adopted equal to 0.04 for iodine and 0.01 for other elements, except for noble gases and C14 carbon (for which no filtering was assumed).
- During normal operation, the isotopes contributing to the highest release values for the TeResa reactor are isotopes from the group of noble gases and iodine.
- The values of releases with the use of filtration systems on an annual basis are below the release limits (used in the operation of the MARIA reactor, it is respectively $1.19 \cdot 10^{12}$ [Bq / y] (noble gases), $6.57 \cdot 10^{7}$ [Bq / y] (argon – Ar41) which is 0.867% of the total release limit for noble gases and argon and $1.39 \cdot 10^{9}$ [Bq / y] for iodine isotopes, i.e. 27% of the release limit.
- The iodine release value is overestimated due to conservative assumptions about the release of iodine from the fuel during normal operation. In normal operation and with fuel operating conditions below 1200 ° C, TRISO fuel shows very good retention characteristics of fission products (including iodine). The predicted actual iodine release value will be lower and the use of a calculation code to estimate fuel releases is recommended in the future.
- Comparison to the INL methodology confirms the conservativeness of used attentuation factor for the system.

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Dziękuję za uwagę!

eleonora.skrzypek@ncbj.gov.pl

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