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TRISO Fuel: Mechanistic Source Term

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TRISO Fuel: Mechanistic Source Term

Advanced Reactor Technologies

Idaho National Laboratory

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NRC HTGR Training April 16-17, 2019

Idaho National Laboratory

ADVANCED REACTOR TECHNOLOGIES

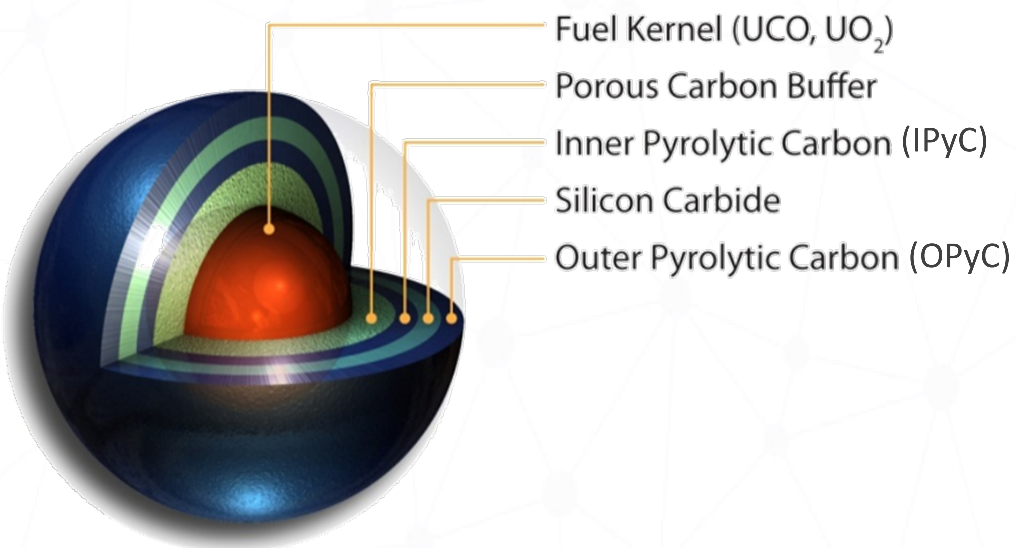
Outline

- Barriers to radionuclide (RN) release in high-temperature gas-cooled reactors (HTGRs)
- Radionuclide Design criteria
- Computational tools to predict radionuclide release
- Simple model to estimate source term from HTGRs

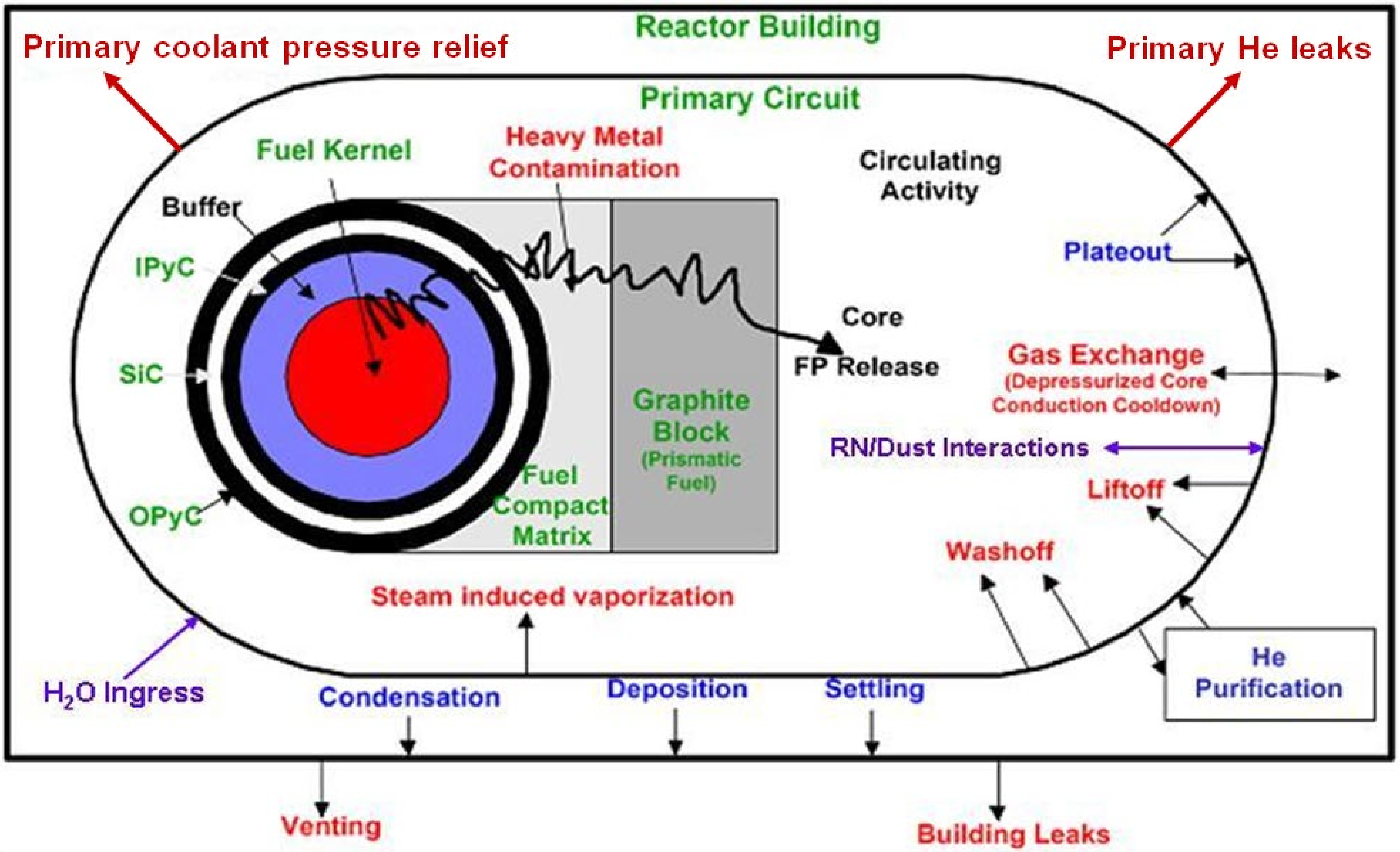
The Training Course delivered to the NRC in 2010 included a module discussing radionuclide behavior in HTGRs (Module 13). You are encouraged to review that course material for additional details.

Radionuclide Barriers

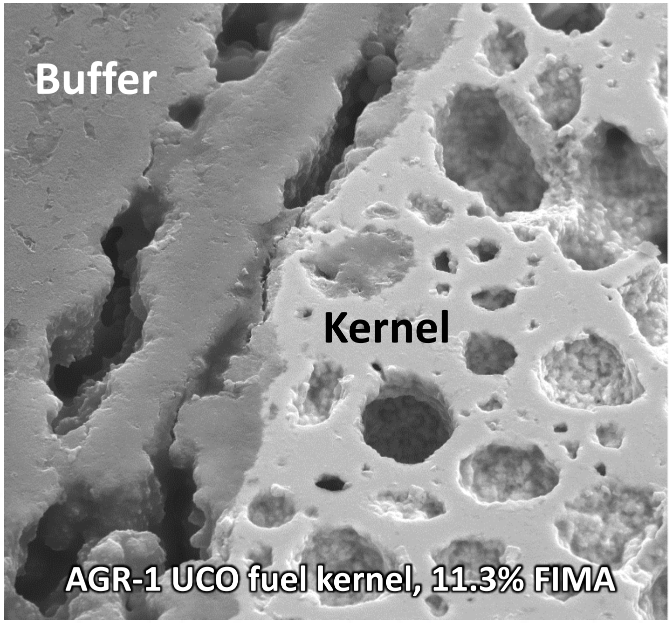
- HTGR designs employ multiple radionuclide release barriers
  - Fuel kernels
  - Particle coatings (most important barrier)
  - Fuel-element matrix and fuel-element graphite (prismatic reactor)
  - Primary coolant pressure boundary
  - Reactor building (RB)
- These multiple radionuclide barriers provide defense in depth



HTGR Radionuclide Sources and Pathways



Radionuclide Release Barrier: Fuel Kernel



- Potential release mechanisms
  - Fission recoil
  - Diffusion
  - Hydrolysis (reaction with H_2O)
- Controlling parameters
  - Fuel temperatures
  - Time
  - H_2O concentration
  - Burnup
- Barrier performance
  - Fractional gas release function of time/temperature history
  - Increased gas release in case of hydrolysis
  - Partial diffusive release of volatile fission metals (Ag, Cs > Eu, Sr)
  - Other radionuclides, including actinides, very well retained
  - UO_2 expected to stabilize certain elements to a greater extent than UCO (Sr, Eu)

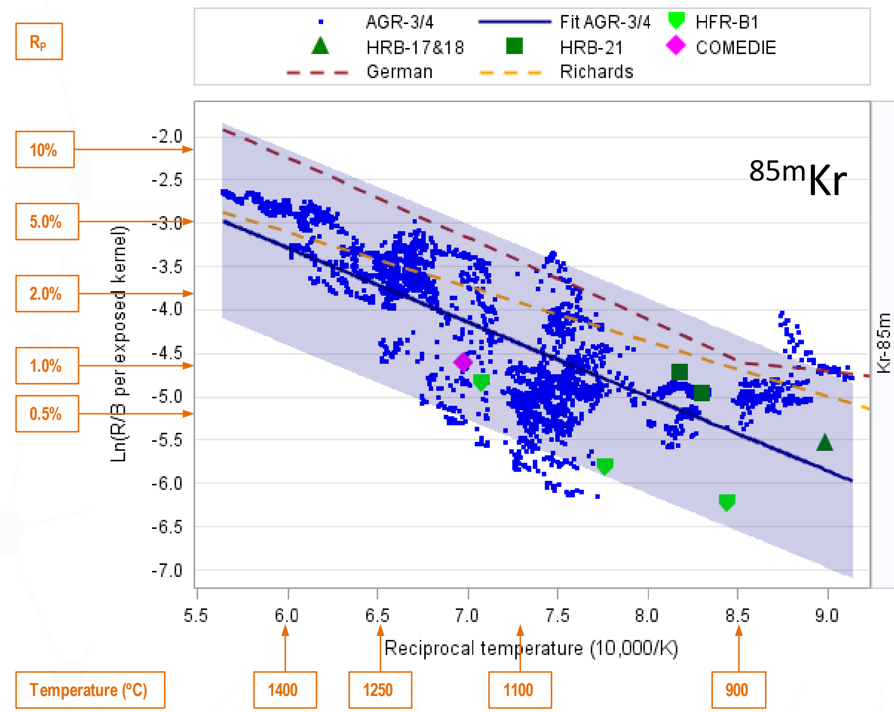
Fission Gas Release

- Short-lived fission gas release in-pile evaluated using the “release-rate-to-birth-rate” (R/B) ratio
- Release is a function of element (Kr, Xe), isotope half life and fuel temperature
- Gas release calculated using a Booth model

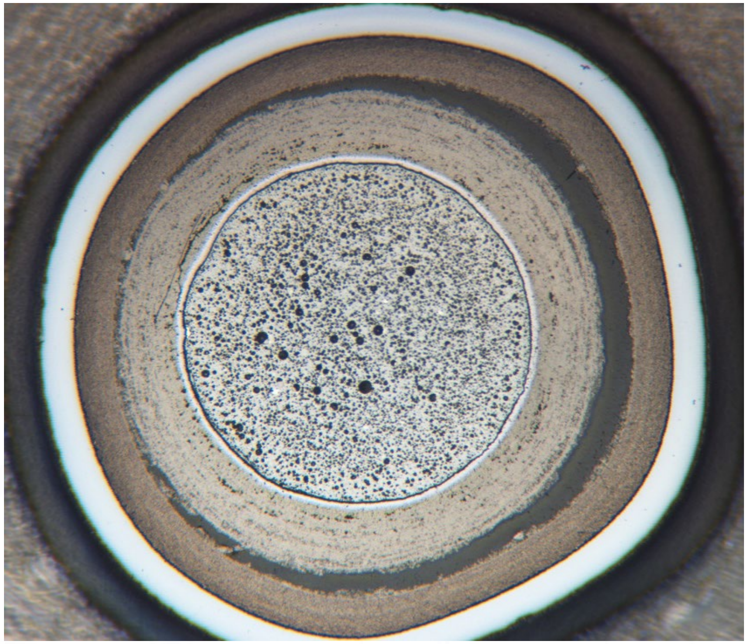
$$R_p \approx \frac{3}{x} = 3\left[\frac{D_0 e^{-\frac{E_a}{RT}}}{\lambda a^2}\right]^n$$

- Various experiments have been performed to determine the R/B ratio for exposed kernels (including recent AGR-3/4 irradiation)
- Isotopes commonly measured:

Kr-85m	Xe-131m
Kr-87	Xe-133
Kr-88	Xe-135
Kr-89	Xe-135m
Kr-90	Xe-137
	Xe-138
	Xe-139



Radionuclide Release Barriers: Particle Coatings



- Potential release mechanisms
  - Diffusion through intact coatings
  - As-fabricated coating defects
  - In-service coating failure
- Controlling parameters
  - Fuel temperatures
  - Time
  - Fast neutron fluence
- Barrier performance
  - Transport through intact coatings:
    - Ag significantly released
    - Other fission products (e.g., Sr, Eu) exhibited more modest release
  - Gases retained by OPyC with defective/failed SiC
  - Metals released when SiC fails
  - TRISO failure rates are very low in modern TRISO fuel

Radionuclide Release Barriers: Matrix/graphite

- Potential release mechanisms
  - Diffusion/vaporization
  - Matrix/graphite oxidation
- Controlling parameters
  - Temperature
  - Time
  - Fast neutron fluence
  - Oxidant concentration
- Barrier performance
  - Sr and Eu exhibit fairly high retention
  - Cs and Ag exhibit lower levels of retention
  - Kr, Xe, and I not retained
  - Sorbed metals assumed to be released by oxidation

Radionuclide Release Barriers: Primary Coolant Circuit

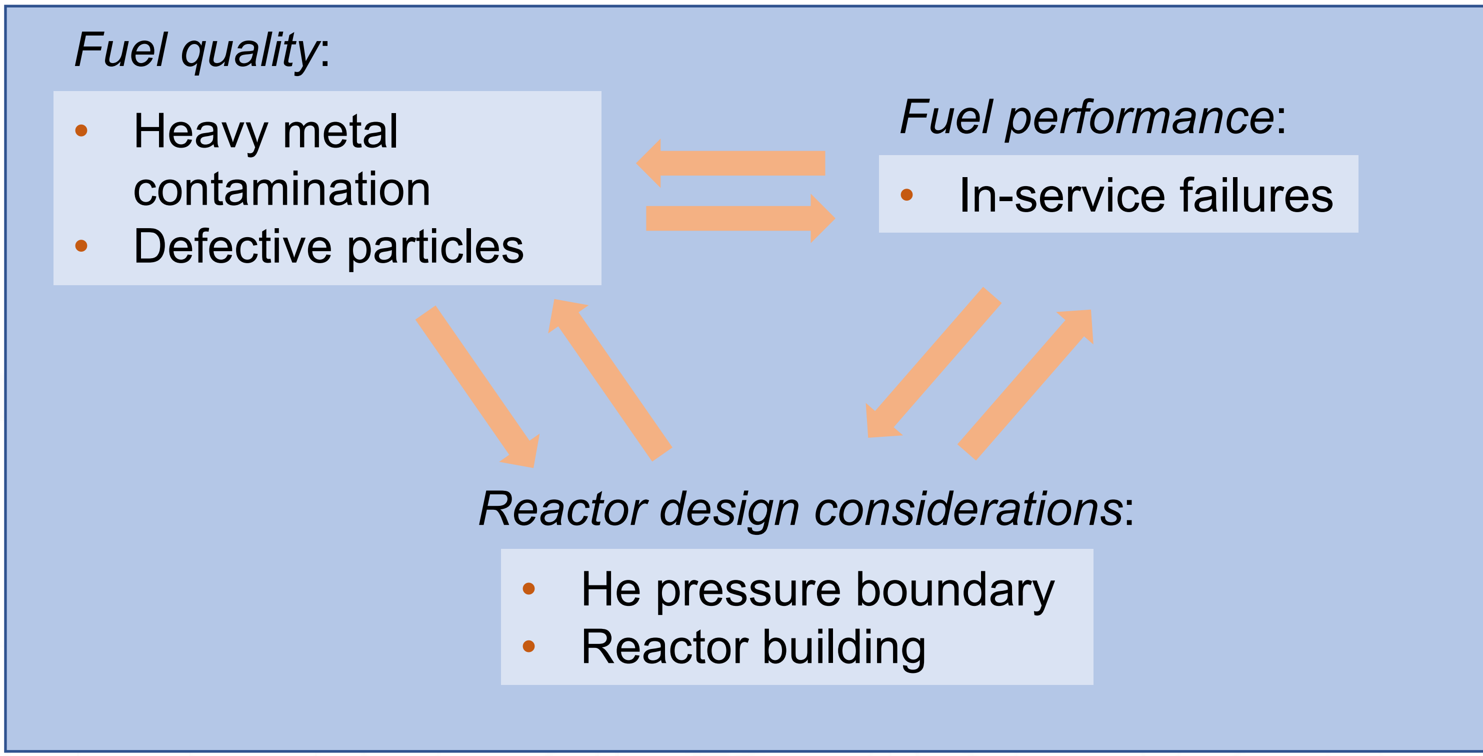
- Potential release mechanisms
  - Primary coolant leaks
  - Liftoff (mechanical re-entrainment)
  - Primary coolant pressure relief
  - Steam-induced vaporization
  - Washoff (removal by liquid H_2O)
- Controlling parameters
  - Temperatures in primary circuit
  - Size/location of coolant leaks
  - Particulate matter in primary circuit
  - Steam/Liquid H_2O ingress and egress
- Barrier performance
  - Condensable radionuclides plate out during normal operation
  - Circulating Kr, Xe and H-3 limited by He purification system
  - Plateout largely retained during rapid blowdowns
  - Radionuclide holdup due to thermal contraction of gas in vessel

Radionuclide Release Barriers: Reactor Building

- Potential release mechanisms
  - Venting through louvers
  - Building leakage
- Controlling parameters
  - Leak path(s) and rates
  - Contaminated steam/liquid H_2O
  - Contaminated particulate matter
  - Temperatures along leak path(s)
- Barrier performance
  - Noble gases decay during holdup
  - Condensable fission products, including iodine, deposit
  - Contaminated steam condenses
  - Contaminated dust settles out and deposits

Radionuclide Release Tradeoffs

- Tradeoffs exist between the relative allocations of the performance of some of the barriers



- Example: allowable fuel contamination levels may be higher if in-service failures are much lower than expected

Particulate Matter (“Dust”) in Primary Circuit May Alter Fission Product Transport Behavior

- Potential sources of dust in HTGRs
  - Foreign material from initial construction or refueling
  - Abrasion/attrition of spherical fuel elements (pebble bed)
  - Erosion or corrosion of fuel or reflector blocks (prismatic)
  - Foreign material from interfacing systems (e.g., HPS)
  - Spallation of friable metallic surface films
  - Carbon deposition from CO decomposition
- Potential impact on fission product (FP) transport
  - Altered FP plateout distributions in primary circuit
  - Enhanced FP release from primary circuit into reactor building
  - Altered FP transport behavior in reactor building
- Experience:
  - Prismatic: very little dust formation; minor impact (PB1, FSV, HTTR)
  - Pebble bed: Measured and characterized in AVR; had impact on plant D & D

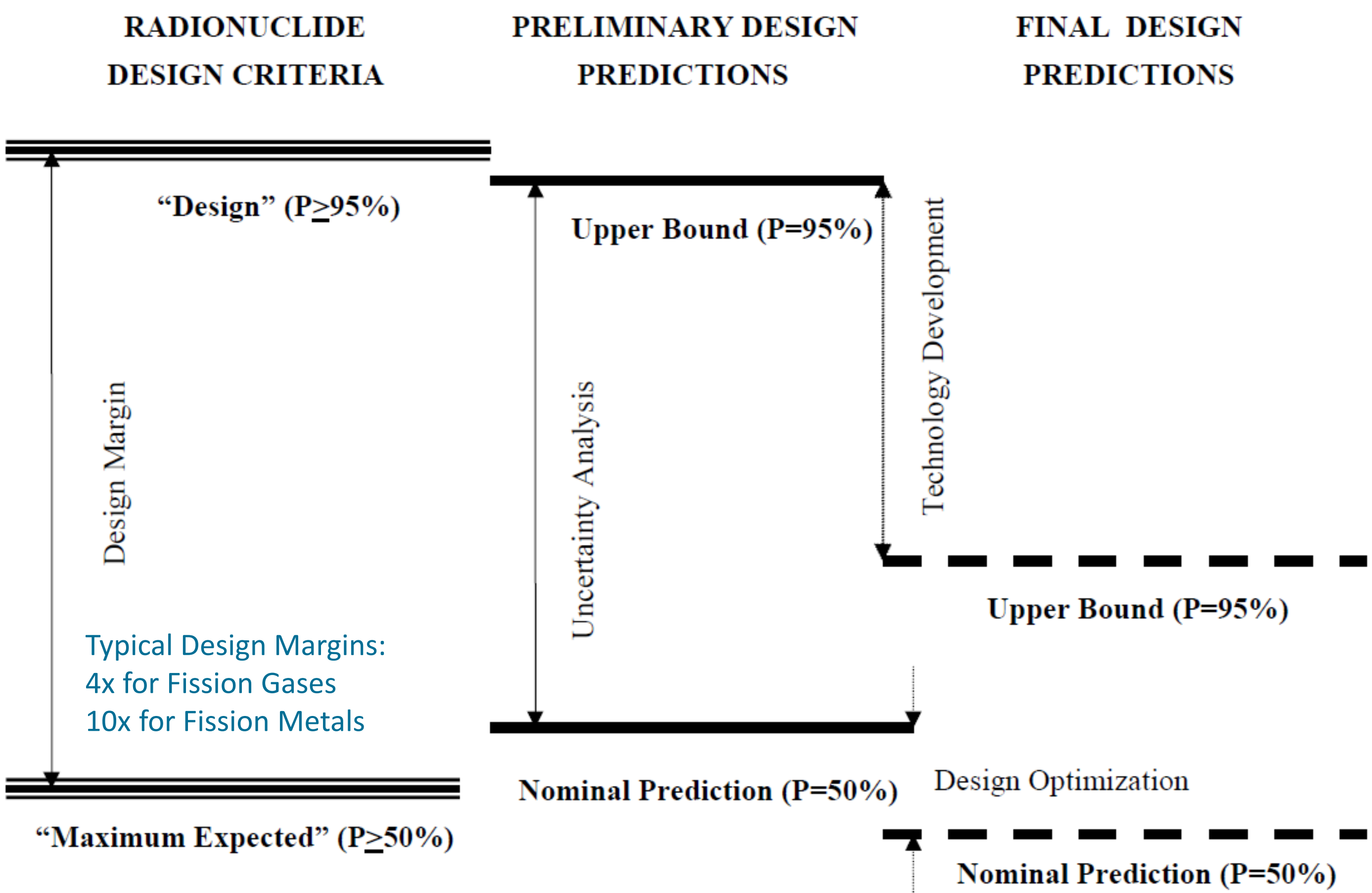
Tritium Release in HTGRs

- Tritium (H-3) will be produced by nuclear reactions
  - Ternary Fission (Yield = ~1×10^{-4})
  - Neutron activation of impurities (He-3 in coolant; Li in graphite)
  - Neutron capture in boron control materials
- Some H-3 will accumulate in primary helium
  - Controlled by He Purification System
  - Significant sorption on core graphite
- Fraction of circulating H-3 in He will permeate through intermediate heat exchanger (IHX) and SG with potential to contaminate process gases and steam
  - Generally not a concern with 750°C outlet temperatures; becomes more important at outlet of 900°C
- H-3 will contribute to public and occupational exposures
  - Environmental releases from plant (liquid discharge)
  - Contaminated products (e.g., hydrogen, bitumen, etc.)
- Data on H-3 transport in reactors and in relevant materials have been obtained with dedicated experiments and through reactor operating experience
- In operating reactors, offsite H-3 release has been below regulatory limits

Radionuclide Design Criteria

- "Top down" approach used to determine allowable radionuclide releases within the functional containment system
- Start with imposed requirements (e.g., site-boundary dose limits)
- Allowable radionuclide inventories in primary circuit derived from radionuclide control requirements
  - Two-tier set of "Radionuclide design criteria" defined to explicitly include safety factors in plant design

Design Margins (Safety Factors) Are Explicitly Included in Radionuclide Design Criteria (Prismatic Example)



A Review of Radionuclide Release from HTGR Cores During Normal Operation, EPRI report 1009382 (2003)

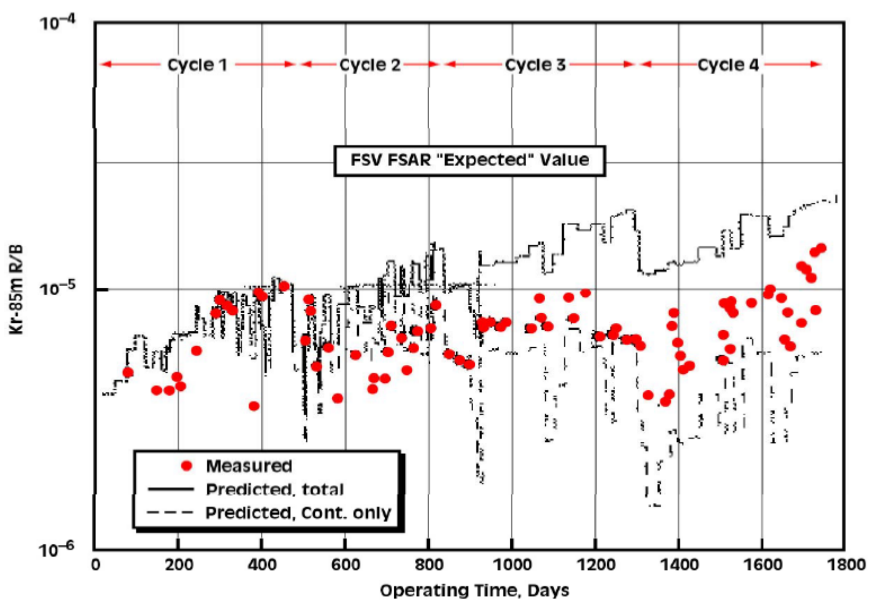
Computational Tools to Predict Radionuclide Behavior in HTGRs

- Design methods for predicting FP transport in HTGRs derived from experimental data
  - Typically, design codes model multiple radionuclide release barriers
  - Core analysis codes typically model fuel performance as well
  - Core codes are typically design specific (i.e., prismatic or pebble)
  - Phenomenological component models derived from data
  - Material property data (e.g., diffusivities, etc.) required as input
- Many comparisons of code predictions with experimental data
  - Reactor surveillance, in-pile tests, etc.
  - Agreement between predictions and measurements has been reasonably good, with predictions somewhat conservative relative to measurements
  - Codes not completely verified and validated

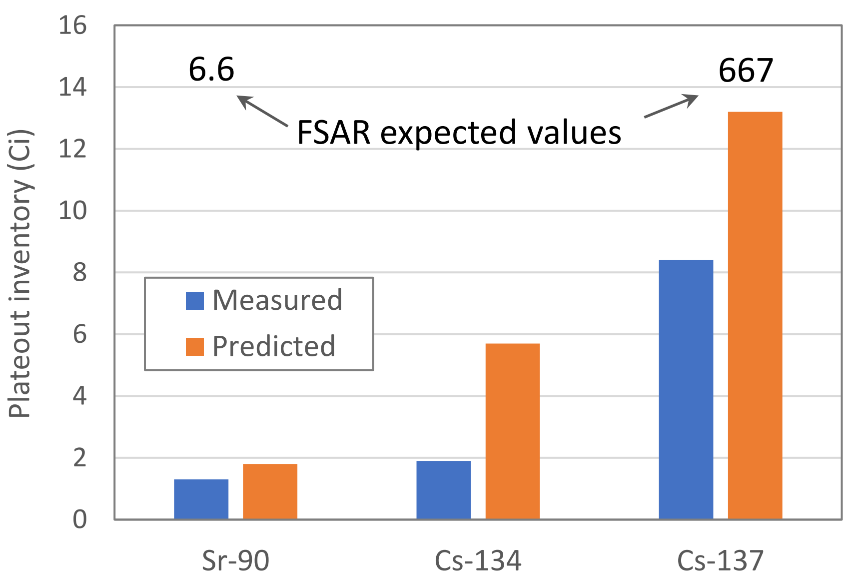
Comparison of Code Predictions with Data

- Past radionuclide release predictions have been reasonably accurate and conservative

Comparison of FSV predicted and measured Kr-85m release



Comparison of FSV predicted and measured fission metal release

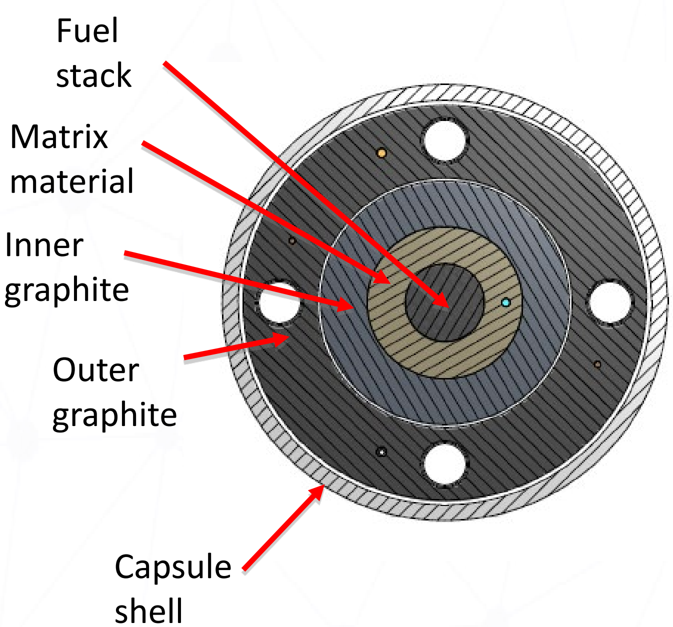


Many additional examples presented in TECDOC-978:

Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997)

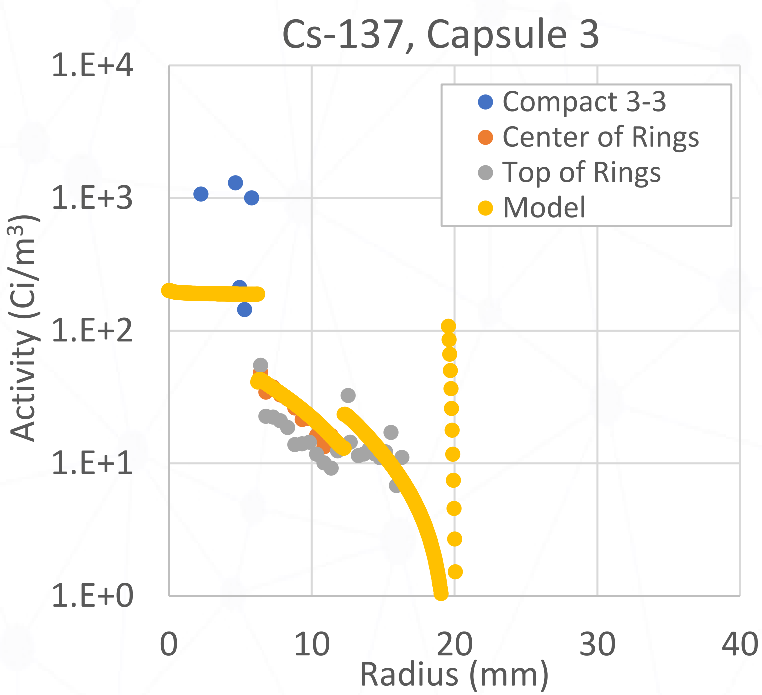
Dedicated Experiments to Refine Source Term Input Parameters

- US Advanced Gas Reactor (AGR) Program AGR-3/4 Irradiation Experiment
- Measure fission product transport in matrix and graphite materials



AGR-3/4 Capsule Cross Section

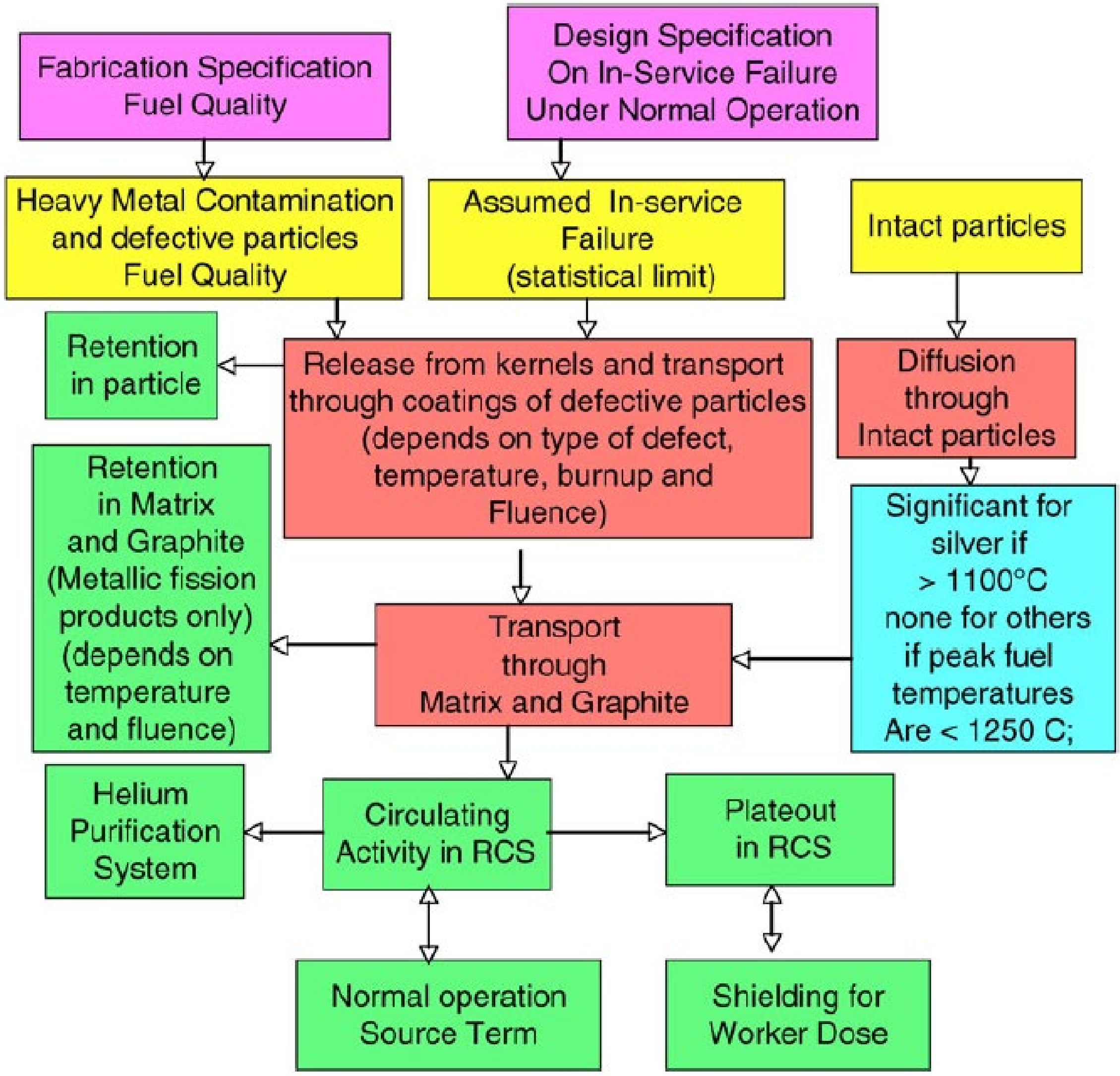
Comparison of measured and predicted Cs profile in AGR-3/4 graphite rings



Simple Model to Calculate Source Terms and Evaluate Barrier Performance

- D.A. Petti et al., Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors, Nucl. Tech. 184 (2013) 181-197
- Radionuclide release from core during normal operation from 4 sources:
  - Release from heavy metal contamination
  - Release from TRISO fuel with SiC defects
  - Release from in-service particle failures
  - Diffusive release through fuel particle coatings
- Total radionuclide inventory in the fuel calculated (e.g., ORIGEN)
- “Attenuation factors” (AFs) are applied at various levels to account for retention of radionuclides (calculated at 50% and 95% confidence)
  - Kernels
  - Coatings
  - Graphite
  - He pressure boundary (liftoff)
  - Reactor building
- AFs determined based on expert opinion (informed by experimental data, calculations)
- Accommodate uncertainties in defect/failure fractions and AFs using Monte Carlo approach

mHTGR Source Term During Normal Operation



D.A. Petti et al., *Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Calculating Inventory Released from Separate Sources

Release from heavy metal contamination (includes contamination outside SiC layers and exposed kernels)

$$R^{i}_{HMC} = \frac{Inv^{i} * HMC}{AF^{i}_{HMC} * AF^{i}_{G}}$$

Release from particles with defective SiC and in-service failure

$$R^{i}_{DSiC+ISF} = \frac{Inv^{i} * (DSiC + ISF)}{AF^{i}_{K} * AF^{i}_{G}}$$

HMC	= level of heavy metal contamination
Inv^{i}	= inventory of fission product *i*
R^{i}_{HMC}	= release of fission product *i* from HMC (Ci)
$R^{i}_{DSiC+ISF}$	= release of fission product *i* from SiC defects and in-service failures (Ci)
AF^{i}_{HMC}	= attenuation factor of fission product *i* for HMC
AF^{i}_{G}	= attenuation factor of fission product *i* in graphite
AF^{i}_{K}	= attenuation factor of fission product *i* in kernel
$DSiC$	= level of SiC defects
ISF	= level of in-service failures

- Similar equations for release through intact particles, inventory retained in the graphite, and plate-out inventory

Radionuclide Release Attenuation Factors

Prismatic reactor barrier core-average attenuation factors during normal operations (700°C Reactor Outlet Temperature)

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%
Noble Gases	10	3	50	17	10^8	10^7	1	1	1	1
I, Br, Se, Te	10	3	50	17	10^8	10^7	1	1	10^6	10^5
Cs, Rb	1	1	3	1	10^8	10^6	5	2	10^6	10^5
Sr, Ba, Eu	1	1	50	20	10^3	200	10^3	300	10^6	10^5
Ag, Pd	1	1	2	1	500[a]	100[a]	2	1	10^6	10^5
Sb	1	1	2	1	10^8	10^6	20	2	10^6	10^5
Mo, Ru, Rh, Tc	1	1	500	30	10^8	10^7	10^3	300	10^6	10^5
La, Ce	1	1	500	30	10^8	10^7	10^3	300	10^6	10^5
Pu, actinides	1	1	10^3	100	10^8	10^7	10^4	10^3	10^6	10^5

[a] Values presented here for Ag-110m. For Ag-111, 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.

D.A. Petti et al., *Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Radionuclide Inventories During Normal Operation: Representative Examples for Prismatic Reactor Design

Mean values for I-131, Cs-137, and Sr-90 inventories (curies) released to the helium pressure boundary and retained in the fuel matrix and graphite

Reactor Design Configuration	I-131		Cs-137		Sr-90	
	In Fuel Matrix and Graphite	In He Pressure Boundary	In Fuel Matrix and Graphite	In He Pressure Boundary	In Fuel Matrix and Graphite	In He Pressure Boundary
600 MW(t) Prismatic 700°C ROT	nil	30	24	5	2750	0.1
600 MW(t) Prismatic 900°C ROT	nil	74	226	254	5680	31

ROT: Reactor outlet temperature

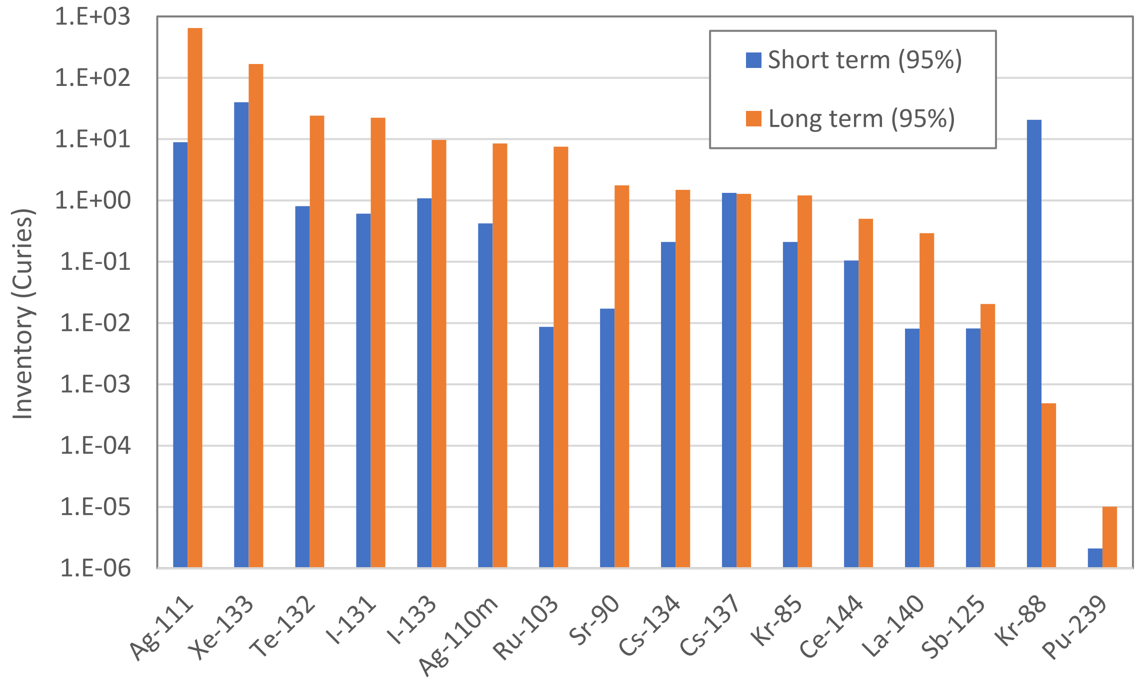
D.A. Petti et al., *Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Evaluating Radionuclide Release During Licensing Basis Events

- Similar approach taken for accident scenarios to determine total source term released from reactor building
- Radionuclide sources in accidents
  - Release from heavy metal contamination
  - Release from TRISO fuel with SiC defects
  - Release from in-service particle failures
  - Diffusive release through fuel particle coatings
  - Inventory in the graphite/matrix from normal operation
  - Lift-off of inventory plated out on the coolant boundary during normal operation
- Different accident scenarios have specific attenuation factors depending on accident conditions (temperature, dry/wet conditions, etc.)
  - e.g., moisture can increase release from exposed kernels and lift-off from the pressure boundary
- Separate calculations for short- and long-term release for accidents (driven by differing half-lives of radioisotopes)

Radionuclide Source Terms During LBEs: Representative Example for Prismatic Reactor Design

Source terms for a break in He pressure boundary, upper 95% confidence levels
600 MW(t) Prismatic 700°C reactor outlet temperature



D.A. Petti et al., *Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Summary

- HTGR designs employ multiple radionuclide release barriers to meet radionuclide control requirements
- Radionuclide transport in HTGRs has been extensively investigated
- Design methods available to predict performance of the radionuclide release barriers during normal operation and accidents
  - Codes have been used extensively for reactor design and analysis, including operating HTGRs
- Many comparisons of code predictions with data
  - Reactor surveillance, in-pile tests, etc.
  - Codes not completely verified and validated
- Additional data from ongoing programs (e.g., US DOE AGR program) will help refine transport parameters and reduce uncertainties
- Contemporary analyses indicate that radionuclide releases during accidents are within acceptable regulatory limits

Suggested Reading

- 2010 HTGR Technology Course for the Nuclear Regulatory Commission
- A Review of Radionuclide Release from HTGR Cores During Normal Operation, EPRI report 1009382 (2003)
- D.A. Petti et al., Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors, Nucl. Tech. 184 (2013) 181-197
- Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997)
- High Temperature Gas Cooled Reactor Fuels and Materials, IAEA, TECDOC-1645 (2010)
- Advances in High Temperature Gas Cooled Reactor Fuel Technology, IAEA, TECDOC-1674 (2012)

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